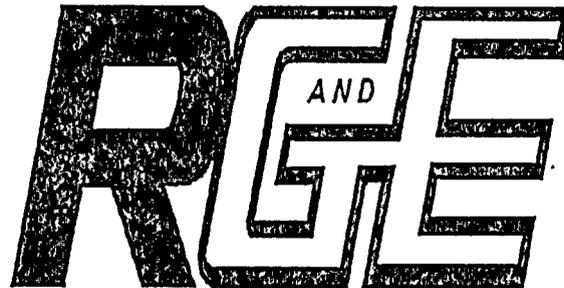


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Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

Attachment C
Chapters 3.5 - 5.0

Volume III

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Two ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1600 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce pressurizer pressure to \leq 1600 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator motor operated isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 1126 cubic feet (50%) and ≤ 1154 cubic feet (82%).	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 700 psig and ≤ 790 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2100 ppm and ≤ 2600 ppm.	31 days on a STAGGERED TEST BASIS
SR 3.5.1.5	Verify power is removed from each accumulator motor operated isolation valve operator when pressurizer pressure is > 1600 psig.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - MODES 1, 2, and 3

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

- NOTES-----
1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. Power may be restored to motor operated isolation valves 878B and 878D for up to 12 hours for the purpose of testing per SR 3.4.14.1 provided that power is restored to only one valve at a time.
 2. Operation in MODE 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of both RCS cold legs exceeds 375°F, whichever comes first.
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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One train inoperable.</p> <p><u>AND</u></p> <p>At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.</p>	<p>A.1 Restore train to OPERABLE status.</p>	<p>72 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Be in MODE 4.	12 hours
C. Two trains inoperable.	C.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.2.1 Verify the following valves are in the listed position. <u>Number</u> <u>Position</u> <u>Function</u> 825A Open RWST Suction to SI Pumps 825B Open RWST Suction to SI Pumps 826A Closed BAST Suction to SI Pumps 826B Closed BAST Suction to SI Pumps 826C Closed BAST Suction to SI Pumps 826D Closed BAST Suction to SI Pumps 851A Open Sump B to RHR Pumps 851B Open Sump B to RHR Pumps 856 Open RWST Suction to RHR Pumps 878A Closed SI Injection to RCS Hot Leg 878B Open SI Injection to RCS Cold Leg 878C Closed SI Injection to RCS Hot Leg 878D Open SI Injection to RCS Cold Leg 896A Open RWST Suction to SI and Containment Spray 896B Open RWST Suction to SI and Containment Spray	12 hours

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.5.2.2 Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3 Verify each breaker or key switch, as applicable, for each valve listed in SR 3.5.2.1, is in the correct position.	31 days
SR 3.5.2.4 Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5 Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.6 Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.7 Verify, by visual inspection, each RHR containment sump suction inlet is not restricted by debris and the containment sump screen shows no evidence of structural distress or abnormal corrosion.	24 months



3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - MODE 4

LCO 3.5.3 One ECCS train shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS Safety Injection (SI) subsystem inoperable.	B.1 Restore required ECCS SI subsystem to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.3.1 -----NOTE----- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ----- SR 3.5.2.4 is applicable for all equipment required to be OPERABLE.</p>	<p>In accordance with applicable SR</p>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST water volume not within limits.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.4.1 Verify RWST borated water volume is \geq 300,000 gallons (88%).	7 days
SR 3.5.4.2 Verify RWST boron concentration is \geq 2300 ppm and \leq 2600 ppm.	7 days

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a large break loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The reactor coolant inventory is vacating the core during this phase through steam flashing and ejection out through the break. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, the core is essentially in adiabatic heatup. The balance of accumulator inventory is available to reflood the core and help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

(continued)

BASES

BACKGROUND
(continued)

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves (841 and 865) are maintained open with AC power removed under administrative control when pressurizer pressure is > 1600 psig. This feature ensures that the valves meet the single failure criterion of manually-controlled electrically operated valves per Branch Technical Position (BTP) ICSB-18 (Ref. 1). This is also discussed in References 2 and 3.

The accumulator size, water volume, and nitrogen cover pressure are selected so that one of the two accumulators is sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that one accumulator is adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for SI signal generation, the diesels starting, and the pumps being loaded and delivering full flow. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and safety injection pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the safety injection pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 5) will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty due to the reduced gas volume. A peak clad temperature penalty is an assumed increase in the calculated peak clad temperature due to a change in an input parameter. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis uses a nominal accumulator volume and includes the line water volume from the accumulator to the check valve due to these competing effects.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the time-frame in which boron precipitation is addressed post LOCA. The maximum boron concentration limit is based on the coldest expected temperature of the accumulator water volume and on chemical effects resulting from operation of the ECCS and the Containment Spray (CS) System. The maximum value of 2600 ppm would not create the potential for boron precipitation in the accumulator assuming a containment temperature of 60°F (Ref. 6). Analyses performed in response to 10 CFR 50.49 (Ref. 7) assumed a chemical spray solution of 2000 to 3000 ppm boron concentration (Ref. 6). The chemical spray solution impacts sump pH and the resulting effect of chloride and caustic stress corrosion on mechanical systems and components. The sump pH also affects the rate of hydrogen generation within containment due to the interaction of CS and sump fluid with aluminum components.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation at 800 psig, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 8 and 9).

The accumulators satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Two accumulators are required to ensure that 100% of the contents of one accumulator will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than one accumulator is injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 5) could be violated.

For an accumulator to be considered OPERABLE, the motor-operated isolation valve must be fully open, power removed above 1600 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1600 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

(continued)

BASES

APPLICABILITY
(continued)

This LCO is only applicable at pressures > 1600 psig. At pressures \leq 1600 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 5) limit of 2200°F.

In MODE 3, with RCS pressure \leq 1600 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood since the accumulator water volume is very small when compared to RCS and RWST inventory. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators are not expected to discharge following a large steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

(continued)

BASES

ACTIONS
(continued)

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of one accumulator cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to ≤ 1600 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both accumulators are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.1

Each accumulator motor-operated isolation valve shall be verified to be fully open every 12 hours. Use of control board indication for valve position is an acceptable verification. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

The borated water volume and nitrogen cover pressure shall be verified every 12 hours for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Main control board alarms are also available for these accumulator parameters. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration shall be verified to be within required limits for each accumulator every 31 days on a STAGGERED TEST Frequency since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day STAGGERED TEST Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is > 1600 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, no accumulators would be available for injection if the LOCA were to occur in the cold leg containing the only OPERABLE accumulator. Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.

REFERENCES

1. Branch Technical Position (BTP) ICSB-18 "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."
 2. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topics VI-7.F, VII-3, VII-6, and VIII-2," dated June 24, 1981.
 3. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
 4. UFSAR, Section 6.3.
 5. 10 CFR 50.46.
 6. UFSAR, Section 3.11.
 7. 10 CFR 50.49.
 8. UFSAR, Section 6.2.
 9. UFSAR, Section 15.6.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - MODES 1, 2, and 3

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

- a. Loss of coolant accident (LOCA) and coolant leakage greater than the capability of the normal charging system;
- b. Rod ejection accident;
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
- d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs and reactor vessel upper plenum. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sump has enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to Containment Sump B for recirculation. After approximately 20 hours, simultaneous ECCS injection is used to reduce the potential for boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

(continued)

BASES

BACKGROUND
(continued)

The ECCS flow paths which comprise the redundant trains consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the RHR pumps, heat exchangers, and the SI pumps. The RHR subsystem consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. The SI subsystem consists of three redundant, 50% capacity pumps which supply two RCS cold leg injection lines. Each injection line is capable of providing 100% of the flow required to mitigate the consequences of an accident. These interconnecting and redundant subsystem designs provide the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, suction headers supply water from the RWST to the ECCS pumps. A common supply header is used from the RWST to the safety injection (SI) and containment spray (CS) System pumps. This common supply header is provided with two in-series motor-operated isolation valves (896A and 896B) that receive power from separate sources for single failure considerations. These isolation valves are maintained open with DC control power removed via a key switch located in the control room. The removal of DC control power eliminates the most likely causes for spurious valve actuation while maintaining the capability to manually close the valves from the control room during the recirculation phase of the accident (Ref. 1). The SI pump supply header also contains two parallel motor-operated isolation valves (825A and 825B) which are maintained open by removing AC power. The removal of AC power to these isolation valves is an acceptable design against single failures that could result in undesirable component actuation (Ref. 2).

(continued)

BASES

BACKGROUND
(continued)

A separate supply header is used for the residual heat removal (RHR) pumps. This supply header is provided with a check valve (854) and motor operated isolation valve (856) which is maintained open with DC control power removed via a key switch located in the control room. The removal of DC control power eliminates the most likely causes for spurious valve actuation while maintaining the capability to manually close the valve from the control room during the recirculation phase of the accident (Ref. 3).

The three SI pumps feed two RCS cold leg injection lines. SI Pumps A and B each feeds one of the two injection lines while SI Pump C can feed both injection lines. The discharge of SI Pump C is controlled through use of two normally open parallel motor operated isolation valves (871A and 871B). These isolation valves are designed to close based on the operating status of SI Pumps A and B to ensure that SI Pump C provides the necessary flow through the RCS cold leg injection line containing the failed pump.

The discharges of the two RHR pumps and heat exchangers feed a common injection line which penetrates containment. This line then divides into two redundant core deluge flow paths each containing a normally closed motor operated isolation valve (852A and 852B) and check valve (853A and 853B) which provide injection into the reactor vessel upper plenum.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the steam generators provide core cooling until the RCS pressure decreases below the SI pump shutoff head.

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to Containment Sump B (Refs. 4 and 5). This transfer is accomplished by stopping the RHR pumps, isolating RHR from the RWST by closing motor operated isolation valve 856, opening the Containment Sump B motor operated isolation valves to RHR (850A and 850B) and then starting the RHR pumps. The SI and CS pumps are then stopped and the RWST isolated by closing motor operated isolation valve 896A and 896B for the SI and CS pump common supply header and closing motor operated isolation valve 897 or 898 for the SI pumps recirculation line.

(continued)

BASES

BACKGROUND
(continued)

The RHR pumps then supply the SI pumps if the RCS pressure remains above the RHR pump shutoff head as correlated through core exit temperature, containment pressure, and reactor vessel level indications (Ref. 6). The RHR pumps can also provide suction to the CS pumps for containment pressure control. This high-head recirculation path is provided through RHR motor operated isolation valves 857A, 857B, and 857C. These isolation valves are interlocked with valves 896A, 896B, 897, and 898. This interlock prevents opening of the RHR high-head recirculation isolation valves unless either 896A or 896B are closed and either 897 or 898 are closed. If RCS pressure is such that RHR provides adequate core and containment cooling, the SI and CS pumps remain in pull-stop. During recirculation, flow is discharged through the same paths as the injection phase. After approximately 20 hours, simultaneous injection by the SI and RHR pumps is used to prevent boron precipitation (Ref. 7). This consists of providing SI through the RCS cold legs and into the lower plenum while providing RHR through the core deluge valves into the upper plenum.

The two redundant flow paths from Containment Sump B to the RHR pumps also contain a motor operated isolation valve located within the sump (851A and 851B). These isolation valves are maintained open with power removed to improve the reliability of switchover to the recirculation phase. The operators for isolation valves 851A and 851B are also not qualified for containment post accident conditions. The removal of AC power to these isolation valves is an acceptable design against single failures that could result in an undesirable actuation (Ref. 2).

The SI subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a steam line break (SLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

(continued)

BASES

BACKGROUND
(continued)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet AIF-GDC 44 (Ref. 8).

APPLICABLE
SAFETY ANALYSIS

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;
- d. Core is maintained in a coolable geometry; and
- e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an SLB event and helps ensure that containment temperature limits are met post accident.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

Both ECCS subsystems are taken credit for in a large break LOCA event at full power (Refs. 6 and 10). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the pumps. The SGTR and SLB events also credit the SI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected by the SI pumps into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core. The RHR pumps inject directly into the core barrel by upper plenum injection.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 10 and 11). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates quickly enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the SI pumps deliver sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and transferring suction to Containment Sump B. This includes securing the motor operated isolation valves as specified in SR 3.5.2.1 in position by removing the power sources as listed below.

<u>EIN</u>	<u>Position</u>	<u>Secured in Position By</u>
825A	Open	Removal of AC Power
825B	Open	Removal of AC Power
826A	Closed	Removal of AC power
826B	Closed	Removal of AC Power
826C	Closed	Removal of AC Power
826D	Closed	Removal of AC Power
851A	Open	Removal of AC power
851B	Open	Removal of AC Power
856	Open	Removal of DC Control Power
878A	Closed	Removal of AC Power
878B	Open	Removal of AC Power
878C	Closed	Removal of AC Power
878D	Open	Removal of AC Power
896A	Open	Removal of DC Control Power
896B	Open	Removal of DC Control Power

The major components of an ECCS train consists of an RHR pump and heat exchanger taking suction from the RWST (and eventually Containment Sump B), and capable of injecting through one of the two isolation valves to the reactor vessel upper plenum and one of the two lines which provide high-head recirculation to the SI and CS pumps.

(continued)

BASES

LCO
(continued)

Also included within the ECCS train are two of three SI pumps capable of taking suction from the RWST and Containment Sump B (via RHR), and injecting through one of the two RCS cold leg injection lines. In the case where SI Pump C is inoperable, both RCS cold leg injection lines must be OPERABLE to provide 100% of the ECCS flow equivalent to a single train of SI due to the location of check valves 870A and 870B.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

In MODE 4, the ECCS requirements are as described in LCO 3.5.3, "ECCS-MODE 4."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-Water Level \geq 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Water Level $<$ 23 Ft."

As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room or by field test personnel. The note also allows an SI isolation MOV to be powered for up to 12 hours for the performance of this testing.

(continued)

BASES

APPLICABILITY
(continued)

As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," may be necessary since the LTOP arming temperature is near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 4, 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Mode 4 core cooling requirements are addressed by LCO 3.4.6, "RCS Loops - Mode 4," and LCO 3.5.3, "ECCS - MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level $<$ 23 Ft."

ACTIONS

A.1

With one train inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 12) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering 100% design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or necessary supporting systems are not available.

(continued)

BASES

ACTIONS

A.1 (continued)

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one active component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

In the case where SI Pump C is inoperable, both RCS cold leg injection lines must be OPERABLE to provide 100% of the ECCS flow equivalent to a single train of SI due to the location of check valves 870A and 870B.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 2) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

B.1 and B.2

If the inoperable train cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1

If both trains of ECCS are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be immediately entered. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Use of control board indication for valve position is an acceptable verification. Misalignment of these valves could render both ECCS trains inoperable. The listed valves are secured in position by removal of AC power or key locking the DC control power. These valves are operated under administrative controls such that any changes with respect to the position of the valve breakers or key locks is unlikely. The verification of the valve breakers and key locks is performed by SR 3.5.2.3. Mispositioning of these valves can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned valve is unlikely.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position in most cases, would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Verification every 31 days that AC or DC power is removed, as appropriate, for each valve specified in SR 3.5.2.1 ensures that an active failure could not result in an undetected misposition of a valve which affects both trains of ECCS. If this were to occur, no ECCS injection or recirculation would be available. Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at a single point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.2.7

Periodic inspections of the containment sump suction inlet to the RHR System ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
2. Branch Technical Position (BTP) ICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."
3. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 42 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant (TAC No. 79829)," dated June 3, 1991.
4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.
5. NUREG-0821.
6. UFSAR, Section 6.3.
7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic IX-4, Boron Addition System, R. E. Ginna," dated August 26, 1981.
8. Atomic Industrial Forum (AIF) GDC 44, Issued for comment July 10, 1967.

(continued)



BASES

REFERENCES
(continued)

9. 10 CFR 50.46.
 10. UFSAR, Section 15.6.
 11. UFSAR, Section 6.2.
 12. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - MODE 4

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - MODES 1, 2, and 3," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS). The RHR subsystem must also be capable of taking suction from containment Sump B to provide recirculation.

APPLICABLE SAFETY ANALYSES

There are no Applicable Safety Analyses which apply to the ECCS in MODE 4 due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident. Therefore, the ECCS operational requirements are reduced in MODE 4. It is understood in these reductions that certain automatic SI actuations are not available. In this MODE, sufficient time is expected for manual actuation of the required ECCS to mitigate the consequences of a DBA. This time is also required since the RHR System may be aligned to provide normal shutdown cooling while the SI System may be isolated from the RCS due to low temperature overpressure protection (LTOP) concerns.

Therefore, only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered for this LCO due to the time available for operators to respond to an accident. The ECCS trains satisfy Criterion 4 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following an accident.

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump. The major components of an ECCS train during MODE 4 consists of an RHR pump and heat exchanger, capable of taking suction from the RWST (and eventually Containment Sump B), and able to inject through one of two isolation valves to the reactor vessel upper plenum. Also included within the ECCS train are one of three SI pumps capable of taking suction from the RWST and injecting through one of two RCS cold leg injection lines. The high-head recirculation flow path from RHR to the SI pumps is not required in the MODE 4 since there is no accident scenario which prevents depressurization to the RHR pump shutoff head prior to depletion of the RWST.

Based on the expected time available to respond to accident conditions during MODE 4, and the configuration of the RHR and SI trains, ECCS components are OPERABLE if they are capable of being reconfigured to the injection mode (remotely or locally) within 10 minutes. This includes taking credit for an RHR pump and heat exchanger as being OPERABLE if they are being used for shutdown cooling purposes. LCO 3.4.12, "LTOP System" contains additional requirements for the configuration of the SI system.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

(continued)

BASES

APPLICABILITY
(continued)

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level $<$ 23 Ft."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR subsystem. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

(continued)

BASES

ACTIONS
(continued)

B.1

With no ECCS SI subsystem OPERABLE, due to the inoperability of the SI pump or flow path from the RWST, the plant is not prepared to provide high pressure response to an accident requiring SI. The 1 hour Completion Time to restore at least one SI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance description from Bases 3.5.2 apply. This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.

REFERENCES

None.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to both trains of the ECCS and the Containment Spray (CS) System during the injection phase of a loss of coolant accident (LOCA) recovery. A common supply header is used from the RWST to the safety injection (SI) and CS pumps. A separate supply header is used for the residual heat removal (RHR) pumps. Isolation valves and check valves are used to isolate the RWST from the ECCS and CS System prior to transferring to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump based on RWST level. Use of a single RWST to supply both trains of the ECCS and CS System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

The RWST is located in the Auxiliary Building which is normally maintained between 50°F and 104°F (Ref. 1). These moderate temperatures provide adequate margin with respect to potential freezing or overheating of the borated water contained in the RWST.

During normal operation in MODES 1, 2, and 3, the safety injection (SI), RHR, and CS pumps are aligned to take suction from the RWST.

The ECCS and CS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions. The recirculation lines for the RHR and CS pumps are directed from the discharge of the pumps to the pump suction. The recirculation lines for the SI pumps are directed back to the RWST.

(continued)

BASES

BACKGROUND
(continued)

When the suction for the ECCS and CS pumps is transferred to the containment sump, the RWST and SI pump recirculation flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the Auxiliary Building and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

- a. The RWST contains sufficient borated water to support the ECCS and CS system during the injection phase;
- b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and CS pumps at the time of transfer to the recirculation mode of cooling; and
- c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in inadequate NPSH for the RHR pumps when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE
SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and CS pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 3). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of LCO 3.5.2, "ECCS-MODES 1, 2, and 3"; LCO 3.5.3, "ECCS-MODE 4"; and LCO 3.6.6, "Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), NaOH, and Containment Post-Accident Charcoal Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the volume required for Reactor Coolant System (RCS) makeup is a small fraction of the available RCS volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is selected such that switchover to recirculation does not occur until sufficient water has been pumped into containment to provide necessary NPSH for the RHR pumps. The minimum boron concentration is an explicit assumption in the steam line break (SLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the evaluation of chemical effects resulting from the operation of the CS System.

For a large break LOCA analysis, the minimum water volume limit of 300,000 gallons and the lower boron concentration limit are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration is used to determine the time frame in which boron precipitation is addressed post LOCA. The maximum boron concentration limit is based on the coldest expected temperature of the RWST water volume and on chemical effects resulting from operation of the ECCS and the CS System. A value ≤ 2600 ppm would not create the potential for boron precipitation in the RWST assuming an Auxiliary Building temperature of 50°F (Ref. 1). Analyses performed in response to 10 CFR 50.49 (Ref. 2) assumed a chemical spray solution of 2000 to 3000 ppm boron concentration (Ref. 1). The chemical spray solution impacts sump pH and the resulting effect of chloride and caustic stress corrosion on mechanical systems and components. The sump pH also affects the rate of hydrogen generation within containment due to the interaction of CS and sump fluid with aluminum components.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

(continued)



BASES (continued)

LCO The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and CS pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume and boron concentration limits established in the SRs.

APPLICABILITY In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS System OPERABILITY requirements. Since both the ECCS and the CS System must be OPERABLE in MODES 1, 2, 3, and 4; the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level \geq 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level $<$ 23 Ft."

ACTIONS

A.1

With RWST boron concentration not within limits, it must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST boron concentration to within limits was developed considering the time required to change the boron concentration and the fact that the contents of the tank are still available for injection.

(continued)

BASES

ACTIONS
(continued)B.1

With the RWST water volume not within limits, it must be restored to OPERABLE status within 1 hour. In this Condition, neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.5.4.1

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and CS System pump operation on recirculation. Since the RWST volume is normally stable and the RWST is located in the Auxiliary Building which provides sufficient leak detection capability, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.4.2

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. UFSAR, Section 3.11.
 2. 10 CFR 50.49.
 3. UFSAR, Section 6.3 and Chapter 15.
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3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	<p>-----NOTE----- SR 3.0.2 is not applicable. -----</p> <p>Perform required visual examinations and leakage rate testing except for containment air lock and containment mini-purge valve testing, in accordance with the Containment Leakage Rate Testing Program.</p>	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	<p>Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.</p>	In accordance with the Containment Tendon Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs on the affected air lock components.
 2. Separate Condition entry is allowed for each air lock.
 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more containment air locks with one containment air lock door inoperable.</p>	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. <p>-----</p>	<p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>	
	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>	
	A.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One or more containment air locks with containment air lock interlock mechanism inoperable.</p>	<p>-----NOTES----- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. -----</p>	
	<p>B.1 Verify an OPERABLE door is closed in the affected air lock.</p>	<p>1 hour</p>
	<p><u>AND</u></p>	
	<p>B.2 Lock an OPERABLE door closed in the affected air lock.</p>	<p>24 hours</p>
	<p><u>AND</u></p>	
<p>B.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----- Verify an OPERABLE door is locked closed in the affected air lock.</p>	<p>Once per 31 days</p>	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	<u>AND</u>	
	C.2 Verify a door is closed in the affected air lock.	1 hour
D. Required Action and associated Completion Time not met.	C.3 Restore air lock to OPERABLE status.	24 hours
	<u>AND</u> D.1 Be in MODE 3. D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Containment Leakage Rate Testing Program</p>
<p>SR 3.6.2.2 Verify only one door in each air lock can be opened at a time.</p>	<p>24 months</p>



3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Boundaries

LCO 3.6.3 Each containment isolation boundary shall be OPERABLE.

-----NOTES-----

1. Not applicable to the main steam safety valves in MODES 1, 2, and 3.
 2. Not applicable to the main steam isolation valves (MSIVs) in MODE 1, and in MODES 2 and 3 with the MSIVs open or *closed and* not deactivated.
 3. Not applicable to the atmospheric relief valves in MODES 1 and 2, and in MODE 3 with the Reactor Coolant System average temperature (T_{avg}) $\geq 500^{\circ}\text{F}$.
-

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Penetration flow path(s), except for Shutdown Purge System valve flow paths, may be unisolated intermittently under administrative controls.
 2. Separate Condition entry is allowed for each penetration flow path.
 3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation boundaries.
 4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation boundary leakage results in exceeding the overall containment leakage rate acceptance criteria.
-

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Only applicable to penetration flow paths which do not use a closed system as a containment isolation boundary. ----- One or more penetration flow paths with one containment isolation boundary inoperable except for mini-purge valve leakage not within limit.</p>	<p>A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.</p> <p><u>AND</u></p> <p>A.2 -----NOTE----- Isolation boundaries in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify the affected penetration flow path is isolated.</p>	<p>4 hours</p> <p>Once per 31 days for isolation boundaries outside containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation boundaries inside containment</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. -----NOTE----- Only applicable to penetration flow paths which do not use a closed system as a containment isolation boundary. -----</p> <p>One or more penetration flow paths with two containment isolation boundaries inoperable except for mini-purge valve leakage not within limit.</p>	<p>B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	<p>1 hour</p>

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. One or more mini-purge penetration flow paths with two valves not within leakage limits.</p>	<p>E.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>E.2 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	
<p>F. Required Action and associated Completion Time not met.</p>	<p>F.1 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u></p> <p>F.2 Be in MODE 5.</p>	<p>36 hours</p>



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.1 Verify each mini-purge valve is closed, except when the penetration flowpath(s) are permitted to be open under administrative control.</p>	<p>31 days</p>
<p>SR 3.6.3.2 -----NOTES----- 1. Isolation boundaries in high radiation areas may be verified by use of administrative controls. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. ----- Verify each containment isolation boundary that is located outside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function except for containment isolation boundaries that are open under administrative controls.</p>	<p>92 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.3 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Isolation boundaries in high radiation areas may be verified by use of administrative means. 2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal. <p>-----</p> <p>Verify each containment isolation boundary that is located inside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation accident function, except for containment isolation boundaries that are open under administrative controls.</p>	<p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days</p>
<p>SR 3.6.3.4 Verify the isolation time of each automatic containment isolation valve is within limits.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.6.3.5 Perform required leakage rate testing of containment mini-purge valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Containment Leakage Rate Program.</p>
<p>SR 3.6.3.6 Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.</p>	<p>24 months</p>



3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ 1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LC0 3.6.5 Containment average air temperature shall be $\leq 120^{\circ}\text{F}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), NaOH, and Containment Post-Accident Charcoal Systems

LCO 3.6.6 Two CS trains, four CRFC units, two post-accident charcoal filter trains, and the NaOH system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CS train inoperable.	A.1 Restore CS train to OPERABLE status.	72 hours
B. One post-accident charcoal filter train inoperable.	B.1 Restore post-accident charcoal filter to OPERABLE status.	7 days
C. Two post-accident charcoal filter trains inoperable.	C.1 Restore one post-accident charcoal filter train to OPERABLE status.	72 hours
D. NaOH system inoperable.	D.1 Restore NaOH System to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p>	<p>E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.</p>	<p>6 hours 84 hours</p>
<p>F. One or two CRFC units inoperable.</p>	<p>F.1 -----NOTE----- Required Action F.1 only required if CRFC unit A or C is inoperable. ----- Declare associated post-accident charcoal filter train inoperable. <u>AND</u> F.2 Restore CRFC unit(s) to OPERABLE status.</p>	<p> Immediately 7 days</p>
<p>G. Required Action and associated Completion Time of Condition F not met.</p>	<p>G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 5.</p>	<p>6 hours 36 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>H. Two CS trains inoperable.</p> <p><u>OR</u></p> <p>NaOH System and one or both post-accident charcoal filter trains inoperable.</p> <p><u>OR</u></p> <p>Three or more CRFC units inoperable.</p> <p><u>OR</u></p> <p>One CS and two post-accident charcoal filter trains inoperable.</p>	<p>H.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.6.1 Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.	In accordance with applicable SRs.
SR 3.6.6.2 Verify each CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.3 Verify each NaOH System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.4 Operate each CRFC unit for \geq 15 minutes.	31 days
SR 3.6.6.5 Verify cooling water flow through each CRFC unit.	31 days
SR 3.6.6.6 Operate each post-accident charcoal filter train for \geq 15 minutes.	31 days
SR 3.6.6.7 Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.8	Verify NaOH System solution volume is \geq 4500 gal.	184 days
SR 3.6.6.9	Verify NaOH System tank NaOH solution concentration is \geq 30% by weight.	184 days
SR 3.6.6.10	Perform required post-accident charcoal filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.6.11	Perform required CRFC unit testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.6.6.12	Verify each automatic CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.13	Verify each CS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.14	Verify each CRFC unit starts automatically on an actual or simulated actuation signal.	24 months
SR 3.6.6.15	Verify each post-accident charcoal filter train damper actuates on an actual or simulated actuation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS: (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.6.16 Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.17 Verify spray additive flow through each eductor path.	5 years
SR 3.6.6.18 Verify each spray nozzle is unobstructed.	10 years

3.6 CONTAINMENT SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombinder inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore hydrogen recombinder to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. <u>AND</u> B.2 Restore one hydrogen recombinder to OPERABLE status.	1 hour <u>AND</u> Once per 12 hours thereafter 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Perform a system functional check for each hydrogen recombiner.	24 months
SR 3.6.7.2	Perform CHANNEL CALIBRATION for each hydrogen recombiner actuation and control channel.	24 months

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete containment structure, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) in accordance with Atomic Industry Forum (AIF) GDC 10 and 49 (Ref. 1). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat base mat, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Each weld seam on the inside of the liner has a leak test channel welded over it to allow independent testing of the liner when the containment is open. The liner is also insulated with closed-cell polyvinyl foam covered with metal sheeting up to the containment spray ring headers. The function of the liner insulation is to limit the mean temperature rise of the liner to only 10°F at the time associated with maximum pressure following a DBA (Ref. 2).

The containment hemispherical dome is constructed of reinforced concrete designed for all DBA related moments, axial loads, and shear forces. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. The base mat is a reinforced concrete slab that is connected to the cylinder wall by use of a hinge design which prevents the transfer of imposed shear from the cylinder wall to the base mat. This hinge consists of elastomer bearing pads located between the bottom of the cylinder wall and the base mat, and high strength steel bars which connect the cylinder walls horizontally to the base mat (Ref. 2).

(continued)

BASES

BACKGROUND
(continued)

The cylinder wall is connected to sandstone rock located beneath the containment by use of 160 post-tensioned rock anchors that are coupled with tendons located in the cylinder wall. This design ensures that the rock acts as an integral part of the containment structure.

The concrete containment structure is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the outside environment to within the limits of 10 CFR 100 (Ref. 3). SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 4), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

- a. All penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE automatic containment isolation system, or
 2. Closed by OPERABLE containment isolation boundaries, except as provided in LCO 3.6.3, "Containment Isolation Boundaries."
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

APPLICABLE
SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 5). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was originally strength tested at 69 psig (115% of design). The acceptance criteria for this test was 0.1% of the containment air weight per day at 60 psig which was based on the construction techniques that were used (Ref. 5). Following successful completion of this test, the accident analyses were performed assuming a leakage rate of 0.2% of the containment air weight per day. This leakage rate, in combination with the minimum containment engineered safeguards operating (i.e., either 2 post-accident charcoal filter trains and no containment spray, 1 post-accident charcoal filter train and 1 containment spray train, or no post-accident charcoal filter trains and 2 containment spray trains) results in offsite doses well within the limits of 10 CFR 100 (Ref. 3) in the event of a DBA.

The leakage rate of 0.2% of the containment air weight per day is defined in 10 CFR 50, Appendix J, Option B (Ref. 4), as L_a : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.2% per day in the safety analysis at $P_a = 60$ psig.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_a$ except prior to entering MODE 4 for the first time following performance of periodic testing performed in accordance with 10 CFR 50, Appendix J, Option B. At that time, the combined Type B and C leakage must be $< 0.6 L_a$ on a maximum pathway leakage rate (MXPLR) basis, and the overall Type A leakage must be $< 0.75 L_a$. At all other times prior to performing as found testing, the acceptance criteria for Type B and C testing is $< 0.6 L_a$ on a minimum pathway leakage rate (MNPLR) basis. In addition to leakiness considerations following a design basis LOCA, containment OPERABILITY also requires structural integrity following a DBA.

Compliance with this LCO will ensure a containment configuration, including personnel and equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and mini-purge valves with resilient seals (LCO 3.6.3) and administrative limits for individual isolation boundaries are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODE 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, the containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and mini-purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes these limits to be exceeded. As left leakage prior to entering MODE 4 for the first time following performance of required 10 CFR 50, Appendix J periodic testing, is required to be $< 0.6 L_a$ for combined Type B and C leakage on a MXPLR basis, and $< 0.75 L_a$ for overall Type A leakage (Ref. 6). At all other times between the required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_a$. This is maintained by limiting combined Type B and C leakage to $< 0.6 L_a$ on a MXPLR basis until performance of as found testing. At $\leq 1.0 L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are generally consistent with the recommendations of Regulatory Guide 1.35 (Ref. 7) except that tendon material tests and inspections are not required (Ref. 8).

(continued)

BASES (continued)

- REFERENCES
1. Atomic Industry Forum, GDC 10 and 49, issued for comment July 10, 1967.
 2. UFSAR, Section 3.8.1.
 3. 10 CFR 100.
 4. 10 CFR 50, Appendix J, Option B.
 5. UFSAR, Section 6.2.
 6. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0.
 7. Regulatory Guide 1.35, Revision 2.
 8. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Safety Evaluation, Containment Vessel Tendon Surveillance Program," dated August 19, 1985.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

There are two containment air locks installed at Ginna Station, an equipment hatch and a personnel hatch. Both air locks are nominally a right circular cylinder with a door at each end to allow personnel access. The two doors on each airlock are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains a double-tongue, single gasketed seal and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide a control board alarm if any door is opened. A single control board alarm exists for all four access doors. Additionally, a control board alarm is provided if high pressure exists between the two doors for either airlock.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analyses.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.2% of containment air weight per day (Ref. 1). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 2), as $L_a = 0.2\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_a = 60$ psig following the design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

The equipment hatch and personnel hatch containment air locks form part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate following a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

(continued)

BASES

LCO
(continued)

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the 10 CFR 50, Appendix J Type B air lock leakage test (i.e., SR 3.6.2.1), and both air lock doors must be OPERABLE such that they are closed with leakage within acceptable limits. The interlock allows only one door of an air lock to be opened at a time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment. Normal entry into and exit from containment does not rendered the airlock inoperable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODE 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE. Therefore, the containment air locks are not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)



BASES (continued)

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment." This evaluation should be initiated immediately after declaring a containment air lock inoperable. This is required since the inoperability of an air lock may result in a significant increase in the overall containment leakage rate.

(continued)

BASES

ACTIONS
(continued)

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. If the between air lock door volume exceeds the allowed leakage criteria, and leakage is verified to be into containment (e.g., leakage through the equalizing valve), then the inner airlock door shall be declared inoperable and this Condition entered. If leakage exists from containment to the outside environment, then Condition C is entered. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour and may consist of verifying the control board alarm status for the airlock doors. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 specifies that Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed and Required Actions C.1, C.2, and C.3 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered to be inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note allows performing other activities (i.e., non TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

(continued)

BASES

ACTIONS
(continued)

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A with the exception that both air lock doors are still OPERABLE and either door can be used to isolate the air lock penetration.

The Required Actions have been modified by two Notes. Note 1 specifies that Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed and Required Actions C.1, C.2, and C.3 are the appropriate remedial actions. Note 2 allows entry into and exit from containment through an air lock with an inoperable air lock interlock mechanism under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS
(continued)

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B (e.g., both doors of an airlock are inoperable), Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within the limits of SR 3.6.2.1. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the inoperable air lock door within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits due to the large margin between the airlock leakage and the containment overall leakage acceptance criteria.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock and the containment overall leakage rate is acceptable.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established based on industry experience. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is as required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 requires that the results of this SR be evaluated against the acceptance criteria of the Containment Leakage Rate Testing Program. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment airlock door is opened, this test is only required to be performed once every 24 months. The 24 month Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES

1. UFSAR, Section 6.2.1.1.
 2. 10 CFR 50, Appendix J, Option B.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Boundaries

BASES

BACKGROUND

The containment isolation boundaries form part of the containment pressure barrier and provide a means for fluid penetrations to be provided with two isolation boundaries. These isolation boundaries are either passive or active (automatic). Manual valves, check valves, de-activated automatic valves secured in their closed position, blind flanges, and closed systems are considered passive boundaries. Automatic valves designed to close without operator action following an accident, are considered active boundaries. Two boundaries in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses in accordance with Atomic Industry Forum (AIF) GDC 53 and 57 (Ref. 1). These active and passive boundaries make up the Containment Isolation System.

The Containment Isolation System is designed to provide isolation capability following a Design Basis Accident (DBA) for all fluid lines which penetrate containment. All major nonessential lines (i.e., fluid systems which do not perform an immediate accident mitigation function) which penetrate containment, except for the main feedwater lines, component cooling water to the reactor coolant pumps, and main steam lines, are either automatically isolated following an accident or are normally maintained closed in MODES 1, 2, 3, and 4. Automatic containment isolation valves are designed to close on a containment isolation signal which is generated by either an automatic safety injection (SI) signal or by manual actuation. The Containment Isolation System can also isolate essential lines at the discretion of the operators depending on the accident progression and mitigation. As a result, the containment isolation boundaries help ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a DBA.

(continued)

BASES

BACKGROUND
(continued)

The OPERABILITY requirements for containment isolation boundaries help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

In addition to the normal fluid systems which penetrate containment, there are two systems which can provide direct access from inside containment to the outside environment.

Shutdown Purge System (36 inch purge valves)

The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access below MODE 4. The supply and exhaust lines each contain one isolation valve and one double gasketed blind flange. Because of their large size, the shutdown purge valves are not qualified for automatic closure from their open position under DBA conditions. Also, due to the design of the blind flange assembly, the isolation valve is not required to be credited as a containment isolation barrier. Therefore, the blind flanges are installed in MODES 1, 2, 3, and 4 to ensure that the containment barrier is maintained (Ref. 2).

Mini-Purge System (6 inch purge valves)

The Mini-Purge System operates to:

- a. Reduce the concentration of noble gases within containment prior to and during personnel access, and
- b. Equalize internal and external pressures.

(continued)

BASES

BACKGROUND

Mini-Purge System (6 inch purge valves) (continued)

The system is designed with supply and exhaust lines both of which contain two air operated isolation valves. Since the valves used in the Mini-Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4; however, emphasis shall be placed on limiting purging and venting times to as low as reasonably achievable.

APPLICABLE
SAFETY ANALYSES

The containment isolation boundary LCO was derived from the assumptions related to minimizing the loss of reactor inventory and establishing the containment barrier during major accidents. As part of the containment barrier, OPERABILITY of devices which act as containment isolation boundaries supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 3). Other DBAs (e.g., locked rotor) result in the release of radioactive material within the reactor coolant system. In the analyses for each of these accidents, it is assumed that containment isolation boundaries are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment and other systems through containment isolation boundaries (including containment mini-purge valves) are minimized. The safety analyses assume that the Shutdown Purge System is isolated at event initiation.

In the calculation of control room and offsite doses following a LOCA (rod ejection accident is assumed to be bounding), the accident analyses assume that 25% of the equilibrium iodine inventory and 100% of the equilibrium noble gas inventory developed from maximum full power operation of the core is immediately available for leakage from containment (Ref. 4). The containment is assumed to leak at the design leakage rate, L_d , for the first 24 hours of the accident and at 50% of this leakage rate for the remaining duration of the accident.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The containment isolation boundaries ensure that the containment design leakage rate remains within L_d by automatically isolating penetrations that do not serve post accident functions and providing isolation capability for penetrations associated with Engineered Safeguards Functions. The maximum isolation time for automatic containment isolation valves is 60 seconds (Ref. 3). This isolation time is based on engineering judgement since the control room and offsite dose calculations are performed assuming that leakage from containment begins immediately following the accident with no credit for transport time or radionuclide decay. The 60 second isolation time takes into consideration the time required to drain piping of fluid which can provide an initial containment barrier before the containment isolation valves are required to close and the conservative assumptions with respect to core damage occurring immediately following the accident. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (only for motor operated valves affected by a loss of offsite power), and containment isolation valve stroke times.

The containment mini-purge valves are air operated valves which have isolation times shorter than 60 seconds since these penetrations may be opened and provide direct access to the outside environment. The accident analyses assume that these valves close prior to a hot rod burst (20 seconds) which occurs following a large break LOCA since the hot rod burst directly leads to higher radiation concentrations within containment. A 5 second isolation time for the mini-purge valves is used for additional conservatism (Ref. 3). The 5 second total isolation response time includes signal delay and containment isolation valve stroke times.

Containment isolation is also required for events which result in hot rod bursts but do not breach the integrity of the RCS (e.g., locked rotor accident). The isolation of containment following these events also isolates the RCS from all non essential systems to prevent the release of radioactive material outside the RCS. The containment isolation time requirements for these events are bounded by those for the LOCA.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The Containment Isolation System is designed to provide two in series boundaries for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds the limits in the safety analyses. This system was originally designed in accordance with AIF GDC 53 (Ref. 1) which does not contain the specific design criteria specified in 10 CFR 50, Appendix A, GDC 55, 56, and 57 (Ref. 5). In general, the Containment Isolation System meets the current GDC requirements; however, several penetrations differ from the GDC from the standpoint of installed valve type (e.g., check valve versus automatic isolation valve) or valve location (e.g., both containment isolation boundaries are located inside containment). The evaluation of these penetrations is provided in Reference 3.

The containment isolation boundaries satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment isolation boundaries form a part of the containment pressure barrier. The containment isolation boundaries' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment barrier leakage rates during a DBA.

The boundaries covered by this LCO are listed in Reference 6. These boundaries consist of isolation valves (manual valves, check valves, air operated valves, and motor operated valves), pipe and end caps, closed systems, and blind flanges. There are three major categories of containment isolation boundaries which are used depending on the type of penetration and the safety function of the associated piping system:

- a. Automatic containment isolation boundaries which receive a containment isolation signal to close following an accident;

(continued)

BASES

LCO
(continued)

- b. Normally closed containment isolation boundaries which are maintained closed in MODES 1, 2, 3, and 4 since they do not receive a containment isolation signal to close and the penetration is not used for normal power operation (but may be used for a long term accident mitigation function); and
- c. Normally open, but nonautomatic containment isolation boundaries which are maintained open since the penetrations are required for normal power operation. Penetrations which utilize these type of isolation boundaries also contain a passive device (i.e., closed system), such that the normally open, but nonautomatic isolation boundary is only closed after the first passive boundary has failed.

The automatic containment isolation boundaries (i.e., valves) are considered OPERABLE when they are capable of closing within the stroke time specified in Reference 6. The normally closed containment isolation boundaries are considered OPERABLE when the manual valves are closed, air operated or motor operated valves are de-activated and secured in their closed position, check valves are closed with flow secured through the valve, blind flanges, pipe and end caps are in place, and closed systems are intact. The normally open, but nonautomatic, containment isolation boundaries (e.g. check valves and manual valves) are considered OPERABLE when they are capable of being closed. In addition, both penetrations associated with the Shutdown Purge System must be isolated by a blind flange containing redundant gaskets, or a single gasketed blind flange with a de-activated automatic isolation valve (i.e., two passive barriers).

Containment isolation boundary leakage per 10 CFR 50, Appendix J, Type B and C testing, is only addressed by LCO 3.6.1, "Containment," and is not a consideration in determination of containment isolation boundary OPERABILITY.

(continued)

BASES

LCO
(continued)

This LCO provides assurance that the containment isolation boundaries will perform their designed safety functions to control leakage from the containment during DBAs.

The LCO is modified by three Notes. The first Note states that the LCO is not applicable to the main steam safety valves in MODES 1, 2, and 3. These valves are addressed by LCO 3.7.1, "Main Steam Safety Valves (MSSVs)," which provides appropriate Required Actions in the event these valves are declared inoperable.

The second Note states that the LCO is not applicable to the main steam isolation valves (MSIVs) in MODE 1, and in MODES 2 and 3 with the MSIVs open or closed and not deactivated. These valves are addressed by LCO 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves."

The third Note states that the atmospheric relief valves are not addressed by this LCO in MODES 1 and 2, and MODE 3 when the Reactor Coolant System average temperature (T_{avg}) is $\geq 500^{\circ}\text{F}$. These valves are addressed by LCO 3.7.4, "Atmospheric Relief Valves (ARVs)," which provides appropriate Required Actions in the event these valves are declared inoperable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODE 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE. Therefore, the containment isolation boundaries are not required to be OPERABLE in MODE 5. The requirements for containment isolation boundaries during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by four Notes. The first Note allows penetration flow paths, except for the Shutdown Purge System valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual qualified in accordance with plant procedures at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the shutdown purge line penetration and the fact that these penetrations exhaust directly from the containment atmosphere to the outside environment, the penetration flow path containing these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation boundary. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation boundaries are governed by subsequent Condition entry and application of associated Required Actions.

A third Note has been added which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation boundary, or as the result of performing the Required Actions described below.

(continued)

BASES

ACTIONS
(continued)

Finally, in the event the isolation boundary leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1. This evaluation should be initiated immediately after declaring a containment isolation boundary inoperable. This is required since the inability of an isolation boundary to close may result in a significant increase in the overall containment leakage rate if the in-series and redundant isolation boundary has a large "as-left" leakage rate associated with it.

A.1 and A.2

In the event one containment isolation boundary in one or more penetration flow paths is inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation boundary that cannot be adversely affected by a single active failure. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the boundary used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

(continued)

BASES

ACTIONS
(continued)

A.1 and A.2

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being isolated following a single failure will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation boundaries outside containment" is appropriate considering the fact that the boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation boundaries and other administrative controls that will ensure that isolation boundary misalignment is an unlikely possibility.

Required Action A.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these boundaries, once they have been verified to be in the proper position, is small.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flowpaths which do not use a closed system as a containment isolation boundary. For those penetrations which do use a closed system, Condition C provides the appropriate actions.

(continued)

BASES

ACTIONS
(continued)

B.1

With two containment isolation boundaries in one or more penetration flow paths inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation boundary that cannot be adversely affected by a single active failure. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. Check valves and closed systems are not acceptable isolation boundaries in this instance since they cannot be assured to meet the design requirements of a normal containment isolation boundary. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.

Following completion of Required Action B.1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action A.2.

Condition B is modified by a Note indicating that this Condition is only applicable to penetration flow paths which do not use a closed system as containment isolation boundary. For those penetrations which do use a closed system, Condition C provides the appropriate actions.

(continued)



BASES

ACTIONS
(continued)

C.1 and C.2

With one or more penetration flow paths with one containment isolation boundary inoperable, the inoperable boundary flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. This Required Action does not require any testing or device manipulation. Rather, it involves verification through a system walkdown, that these isolation boundaries capable of being mispositioned are in the correct position. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of "once per 31 days for isolation boundaries outside containment" is appropriate considering the fact that the boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation boundaries and other administrative controls that will ensure that isolation boundary misalignment is an unlikely possibility.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths which use a closed system as a containment isolation boundary. This Note is necessary since this Condition is written to specifically address those penetration flow paths which utilize a closed system as defined in Reference 7.

Required Action C.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

D.1

In the event one or more containment mini-purge penetration flow paths contain one valve not within the mini-purge valve leakage limits, mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation boundary that cannot be adversely affected by a single active failure. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

(continued)



BASES

ACTIONS
(continued)

D.2

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries and capable of being mispositioned are in the correct position. The Completion Time of "once every 31 days for isolation boundaries outside containment" is appropriate considering the fact that the boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation boundaries and other administrative controls that will ensure that isolation boundary misalignment is an unlikely possibility.

Required Action D.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these boundaries, once they have been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS
(continued)

E.1

In the event one or more containment mini-purge penetration flow paths contain two valves not within the mini-purge valve leakage limits, Required Action E.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current mini-purge results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both mini-purge valves have failed a leakage test or are not within the limits of SR 3.6.3.5. In many instances, containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the mini-purge penetration flow path within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown. In addition, even with both valves failing the leakage test, the overall containment leakage rate can still be within limits due to the large margin between the mini-purge valve leakage and the containment overall leakage acceptance criteria.

E.2

Required Action E.2 requires that the mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated within 1 hour. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action E.2 must have been demonstrated to meet the leakage requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

Following completion of Required Action E.1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action D.2.

(continued)

BASES

ACTIONS
(continued)

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.1

This SR ensures that the mini-purge valves are closed except when the valves are opened under administrative control. The mini-purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, maintenance activities, or for Surveillances that require the valves to be open. To be opened, the valves must be capable of closing under accident conditions, the containment isolation signal to the valves must be OPERABLE, and the effluent release must be monitored to ensure that it remains within regulatory limits. The 31 day Frequency is based on the relative importance of these valves since they provide a direct path to the outside environment and the administrative controls that are in place.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.3.2

This SR requires verification that each containment isolation boundary located outside containment and not locked, sealed or otherwise secured in the required position is performing its containment isolation accident function. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries outside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and a 92 day Frequency to verify their correct position is appropriate. The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time the boundaries are open.

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the isolation times of these valves are verified by SR 3.6.3.4 and the boundaries are required to be OPERABLE.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.3.3

This SR requires verification that each containment isolation boundary located inside containment and not locked, sealed or otherwise secured in the required position and is performing its containment isolation accident function. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation boundaries inside containment and capable of being mispositioned are in the correct position. This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate. The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time they are open.

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the signal provides assurance the valve will be closed following an accident.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.3.4

Verifying that the isolation time of each automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

For containment mini-purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the outside environment), a leakage acceptance criteria of $\leq 0.05 L_a$ when tested at $\geq P_a$ is specified for each mini-purge isolation valve with resilient seals in the Containment Leakage Rate Testing Program. The Frequency of testing is also specified in the Containment Leakage Rate Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. Atomic Industry Forum GDC 53 and 57, issued for comment July 10, 1967.
 2. Branch Technical Position CSB 6-4, "Containment Purging During Normal Operation."
 3. UFSAR, Section 6.2.4 and Table 6.2-15.
 4. Regulatory Guide 1.4, Revision 2.
 5. 10 CFR 50, Appendix A, GDC 55, 56, and 57.
 6. Ginna Station Procedure A-3.3.
 7. NUREG-0800, Section 6.2.4.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, post accident containment pressures could exceed calculated values. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment pressure outside the limits of the LCO violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses performed to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The worst case SLB generates larger mass and energy releases than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case SLB, 59.8 psig, does not exceed the containment design pressure, 60 psig.

The containment was also designed for an external pressure load equivalent to -2.5 psig. However, internal pressure is limited to -2.0 psig based on concerns related to providing continued cooling for the reactor coolant pump motors inside containment.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2). Service Water System (LCO 3.7.8) temperature plays an important role in both maximizing and minimizing containment pressure following a DBA response.

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure. However, the lower pressure limit specified for this LCO is set at a more limiting pressure to ensure continued cooling of the reactor coolant pump motors inside containment which are required to be OPERABLE for a large portion of MODES 1, 2, 3, and 4.

(continued)

BASES (continued)

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 8 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is greater than the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. However, due to the large containment free volume and limited size of the containment Mini-Purge System, 8 hours is allowed to restore containment pressure to within limits. This is justified by the low probability of a DBA during this time period.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that plant operation remains within the limits assumed in the containment analysis. This verification should normally be performed using PI-944. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

Calibration of PI-944 or other containment pressure monitoring devices should be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.
 2. 10 CFR 50, Appendix K.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the Containment Spray (CS) and Containment Recirculation Fan Cooler (CRFC) Systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses to ensure that the total amount of energy within containment is within the capacity of the CS and CRFC Systems. The containment average air temperature is also an important consideration in establishing the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to the capability of the Engineered Safety Feature (ESF) systems to mitigate the accident, assuming the worst case single active failure. Consequently, the ESF systems must continue to function within the environment resulting from the DBA which includes humidity, pressure, temperature, and radiation considerations.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F. This results in a maximum containment air temperature of 374°F.

The initial temperature limit specified in this LCO is also used to establish the environmental qualification operating envelope for containment. The maximum SLB peak containment air temperature was calculated to exist for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses show that the time interval during which the containment air temperature peaked was short enough that the equipment surface temperatures remained below their design temperatures. Also, the equipment and cabling inside containment are protected against the direct effects of a SLB by concrete floors and shields. Therefore, it was concluded that the calculated transient containment air temperature following a LOCA (286°F) becomes limiting for environmental qualification reasons.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum allowable containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured and the OPERABILITY of equipment within containment is maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within the limit within 24 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 24 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

(continued)



BASES

ACTIONS
(continued)

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. There are 6 containment air temperature indicators (TE-6031, TE-6035, TE-6036, TE-6037, TE-6038, and TE-6045) such that a minimum of three should be used for calculating the arithmetic average. The 12 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment temperature condition.

Calibration of these temperature indicators shall be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.
 2. 10 CFR 50.49.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), NaOH, and Containment Post-Accident Charcoal Systems

BASES

BACKGROUND

The CS and CRFC systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the CS System, NaOH System, and the Containment Post-Accident Charcoal System connected to the CRFC units reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The CS, CRFC, NaOH, and Containment Post-Accident Charcoal Systems are designed to meet the requirements of Atomic Industry Forum (AIF) GDC 49, 52, 58, 59, 60, and 61 (Ref. 1). The CS, NaOH, and Post-Accident Charcoal Systems also are designed to limit offsite doses following a DBA within 10 CFR 100 guidelines.

The CRFC System, CS System, NaOH System, and the Containment Post-Accident Charcoal System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained and reduce the potential release of radioactive material, principally iodine, from the containment to the outside environment. The CS System, CRFC System, NaOH System, and the Containment Post-Accident Charcoal System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

(continued)



BASES

BACKGROUND
(continued)

Containment Spray and NaOH Systems

The CS System consists of two redundant, 100% capacity trains. Each train includes a pump, spray headers, spray eductors, nozzles, valves, and piping (see Figure B 3.6.6-1). Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the CS System during the injection phase of operation through a common supply header shared by the safety injection (SI) system. In the recirculation mode of operation, CS pump suction can be transferred from the RWST to Containment Sump B via the residual heat removal (RHR) system.

The CS System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to scavenge fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the CS System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal coolers. However, the CS System can provide additional containment heat removal capability if required. Each train of the CS System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The NaOH mixture is injected into the CS flowpath via a liquid eductor during the injection phase of an accident. The eductors are designed to ensure that the pH of the spray mixture is between 8.3 and 9.1. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid (Ref. 2).

(continued)

BASES

BACKGROUND

Containment Spray and NaOH Systems (continued)

The CS System is actuated either automatically by a containment Hi-Hi pressure signal or manually. DBAs which can generate an automatic actuation signal include the loss of coolant accident (LOCA) and steam line break (SLB). An automatic actuation opens the CS pump motor operated discharge valves (860A, 860B, 860C, and 860D), opens NaOH addition valves 836A and 836B, starts the two CS pumps, and begins the injection phase. A manual actuation of the CS System requires the operator to actuate two separate pushbuttons simultaneously on the main control board to begin the same sequence. The injection phase continues until an RWST low level alarm is received signaling the start of the recirculation phase of the accident.

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to Containment Sump B (Refs. 3 and 4). This transfer is accomplished by stopping the RHR pumps, isolating RHR from the RWST by closing motor operated valve 856, opening the Containment Sump B motor operated isolation valves to RHR (850A and 850B) and then starting the RHR pumps. The SI and CS pumps are then stopped and the RWST isolated by closing motor operated isolation valve 896A or 896B for the SI and CS pump common supply header and closing motor operated isolation valve 897 or 898 for the SI pumps recirculation line.

The RHR pumps then supply the SI pumps if the RCS pressure remains above the RHR pump shutoff head as correlated through core exit temperature, containment pressure, and reactor vessel level indications (Ref. 5). This high-head recirculation path is provided through RHR motor operated isolation valves 857A, 857B, and 857C. These isolation valves are interlocked with 896A, 896B, 897, and 898. This interlock prevents opening of the RHR high head recirculation isolation valves unless either 896A or 896B are closed and either 897 or 898 are closed. If RCS pressure is such that RHR provides adequate injection flow for core cooling, the SI pumps remain in pull-stop.

(continued)

BASES

BACKGROUND

Containment Spray and NaOH Systems (continued)

The CS System is only used during the recirculation phase if containment pressure increases above a pressure at which containment integrity is potentially challenged. Otherwise, the containment heat removal provided by the CRFC units and Containment Sump B (via the RHR system) is adequate to support containment heat removal needs and the limits on sump pH (Refs. 2 and 6).

Operation of the CS System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Recirculation Fan Cooler System

The CRFC System consists of four fan units (A, B, C, and D). Each cooling unit consists of a motor, fan, cooling coils, dampers, moisture separators, high efficiency particulate air (HEPA) filters, duct distributors and necessary instrumentation and controls (see Figure B 3.6.6-2). The moisture separators function to reduce the moisture content of the airstream to support the effectiveness of the post-accident charcoal filters. CRFC units A and D are supplied by one ESF bus while CRFC units B and C are supplied by a redundant ESF bus. All four CRFC units are supplied cooling water by the Service Water (SW) System via a common loop header. Air is drawn into the coolers through the fan and discharged into the containment atmosphere including the various compartments (e.g., steam generator and pressurizer compartments).

During normal operation, at least two fan units are typically operating. The CRFC System, operating in conjunction with other containment ventilation and air conditioning systems, is designed to limit the ambient containment air temperature during normal plant operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

(continued)

BASES

BACKGROUND

Containment Recirculation Fan Cooler System (continued)

In post accident operation following a SI actuation signal, the CRFC System fans are designed to start automatically if not already running. The discharge of CRFC units A and C then transfer to force flow through the post-accident charcoal filters. The temperature of the cooling water supplied by SW System (LCO 3.7.8) is an important factor in the heat removal capability of the fan units.

Containment Post-Accident Charcoal System

The Containment Post-Accident Charcoal System consists of two redundant, 100% capacity trains. Each train includes an airtight plenum containing two banks of charcoal filter cells for removal of radioiodines (see Figure 3.6.6-2). Air flow enters the plenum through two holes in the bottom (one at each end), passes through the charcoal filter banks to the center, and is exhausted from the plenum through a hole in the top. Two normally closed air operated dampers isolate each post-accident charcoal filter train from CRFC units A and C (dampers 5871 and 5872 for Train A and 5874 and 5876 for Train B). A SI signal opens these dampers and closes two bypass dampers from the CRFC units (dampers 5873 for CRFC unit A and 5875 for CRFC unit C) to force flow through the post-accident charcoal filters.

APPLICABLE
SAFETY ANALYSES

The CS System and CRFC System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the LOCA and the SLB which are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the worst case single active failure.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 59.8 psig and the peak containment temperature is 374°F (both experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Temperature," for a detailed discussion.) The analyses and evaluations assume a plant specific power level of 102%, one CS train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 7).

The effect of an inadvertent CS actuation is not considered since there is no single failure, including the loss of offsite power, which results in a spurious CS actuation.

The modeled CS System actuation for the containment analysis is based on a response time associated with exceeding the containment Hi-Hi pressure setpoint to achieving full flow through the CS nozzles. To increase the response of the CS System, the injection lines to the spray headers are maintained filled with water. The CS System total response time of 37.5 seconds (assuming the containment Hi-Hi pressure is reached at time zero) includes diesel generator (DG) startup (for loss of offsite power), opening of the motor operated isolation valves, containment spray pump startup, and spray line filling (Ref. 8).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The modeled CRFC System actuation for the containment analysis is based upon a response time associated with exceeding the SI actuation levels to achieving full CRFC System air and safety grade cooling water flow. The CRFC System total response time of 44 seconds, includes signal delay, DG startup (for loss of offsite power), and service water pump and CRFC unit startup times (Ref. 9).

During a SLB or LOCA, a minimum of two CRFC units and one CS train are required to maintain containment peak pressure and temperature below the design limits.

The CS, NaOH, and Containment Post-Accident Charcoal Systems operate to reduce the release of fission product radioactivity from containment to the outside environment in the event of a DBA. The DBAs that result in a release of radioactive iodine within containment are the LOCA or a rod ejection accident (REA). In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is intact at event initiation to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine released is limited by reducing the iodine concentration present in the containment atmosphere.

The required iodine removal capability of the CS, NaOH, and Containment Post-Accident Charcoal Systems is established by the consequences of the limiting DBA, which is a LOCA. The accident analyses (Ref. 10) assume that either two trains of CS (taking suction from the NaOH System), one CS train and one post-accident charcoal filter train, or two post-accident charcoal filter trains operate to remove radioactive iodine from the containment atmosphere.

The CS System, NaOH System, CRFC System, NaOH System, and the Containment Post-Accident Charcoal System satisfy Criterion 3 of the NRC Policy Statement.

(continued)



BASES (continued)

LCO

During a DBA, a minimum of 2 CRFC units and one CS train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 8). Additionally, two CS trains taking suction from the NaOH System, two CRFC units with post accident charcoal filters (i.e., units A and C), or one CRFC unit with post accident charcoal filters in combination with one CS train are also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two CS trains, four CRFC units, and two post-accident charcoal filter trains and the NaOH System must be OPERABLE. Therefore, in the event of an accident, at least one CS and post-accident charcoal filter train, the NaOH System, and two CRFC units operates, assuming the worst case single active failure occurs.

Each CS train includes a spray pump, spray headers, nozzles, valves, spray eductors, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and manually transferring suction to Containment Sump B via the RHR pumps.

For the NaOH System to be OPERABLE, the volume and concentration of spray additive solution in the tank must be within limits and air operated valves 836A and 836B must be OPERABLE.

Each CRFC unit includes a motor, fan cooling coils, dampers, moisture separators, HEPA filters, duct distributors, instruments, and controls to ensure an OPERABLE flow path. For CRFC units A and C, flow through either the post-accident charcoal filter or the bypass is required for the units to be considered OPERABLE.

Each post-accident charcoal filter train includes a plenum containing charcoal filter banks and isolation dampers to ensure an OPERABLE flow path. CRFC units A and C are also required to be OPERABLE.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the CS System, CRFC System, NaOH System, and the Post-Accident Charcoal System.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the CS System, CRFC System, NaOH System, and the Post-Accident Charcoal System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one CS train inoperable, the inoperable CS train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and CRFC units are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the CS System, the redundant iodine removal afforded by the Containment Post-Accident Charcoal System, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)



BASES

ACTIONS
(continued)

B.1

With one post-accident charcoal filter train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. Each post-accident charcoal filter train is capable of providing 50% of the radioactive iodine removal requirements following a DBA. The loss of CRFC unit A or C also results in its associated post-accident charcoal filter train being inoperable since the post-accident charcoal filter trains do not have their own fan assembly. The 7 day Completion Time of Required Action B.1 to restore the inoperable post-accident charcoal filter train, including the CRFC unit, is justified considering the redundant iodine removal capabilities afforded by the CS and NaOH Systems and the low probability of a DBA occurring during this time period.

C.1

With both post-accident charcoal filter trains inoperable, at least one post-accident charcoal filter train must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time to restore one inoperable post-accident charcoal filter train is justified considering the redundant iodine removal capabilities afforded by the CS System and the low probability of a DBA occurring during this time period. The inoperable post-accident charcoal filter train includes, but is not limited to inoperable CRFC units A and C.

D.1

With the NaOH System inoperable, OPERABLE status must be restored within 72 hours. The 72 hour Completion Time to restore the NaOH System is justified considering the redundant iodine removal capabilities afforded by the Containment Post-Accident Charcoal System and the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

If the inoperable CS train, post-accident charcoal filter trains, or the NaOH System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the inoperable component(s) and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

F.1 and F.2

With one or two CRFC units inoperable, the affected post-accident charcoal filter must be declared inoperable immediately and the inoperable CRFC unit(s) must be restored to OPERABLE status within 7 days. The inoperable CRFC units provided up to 100% of the containment heat removal needs and up to 50% of the iodine removal needs. The 7 day Completion Time is justified considering the redundant heat removal capabilities afforded by combinations of the CS System and CRFC System and the low probability of DBA occurring during this period. If both CRFC units A and C are inoperable, then Condition C must also be entered.

Required Action F.1 is modified by a Note which states that this required action is only applicable if CRFC unit A or C is inoperable. The loss of CRFC unit A or C results in the associated post-accident charcoal filter train being inoperable since the post-accident charcoal filter trains do not have their own fan assembly.

(continued)

BASES

ACTIONS
(continued)

G.1 and G.2

If the Required Action and associated Completion Time of Condition F of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

With two CS trains inoperable, the NaOH System and one or both post-accident charcoal filter trains inoperable, three or more CRFC units inoperable, or one CS and two post-accident charcoal filter trains inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.6.6.1

The applicable SR descriptions from Bases 3.5.2 apply. This SR is required since the OPERABILITY of valves 896A and 896B is also required for the CS System.

SR 3.6.6.2

Verifying the correct alignment for manual, power operated, and automatic valves in the CS flow path provides assurance that the proper flow paths will exist for CS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.3

Verifying the correct alignment for manual, power operated, and automatic valves in the NaOH System flow path provides assurance that the proper flow paths will exist for NaOH System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.4

Operating each CRFC unit for ≥ 15 minutes once every 31 days ensures that all CRFC units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of significant degradation of the CRFC units occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.5

Verifying cooling water (i.e., SW) flow to each CRFC unit provides assurance that the energy removal capability of the CRFC assumed in the accident analyses will be achieved (Ref. 11). The minimum and maximum SW flows are not required to be specifically determined by this SR due to the potential for a containment air temperature transient. Instead, this SR verifies that SW flow is available to each CRFC unit. The 31 day Frequency was developed considering the known reliability of the SW System, the two CRFC train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.6

Operating each post-accident charcoal filter train for ≥ 15 minutes once every 31 days ensures that all trains are OPERABLE and that all dampers are functioning properly. It also ensures that blockage can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the post-accident charcoal filter trains, the redundancy available, and the low probability of significant degradation of the train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.7

Verifying each CS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 12). Since the CS pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice testing confirms component OPERABILITY, trends performance, and detects incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.8

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water that is injected. This SR is performed to verify the availability of sufficient NaOH solution in the spray additive tank. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval since the tank is normally isolated. Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.9

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration since the tank is normally isolated and the probability that any substantial variance in tank volume will be detected.

SR 3.6.6.10

This SR verifies that the required post-accident charcoal filter train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing charcoal adsorber efficiency, minimum system flowrate, and the physical properties of the activated charcoal. The minimum required flowrate through each of the two post-accident charcoal filters is 33,000 cubic feet per minute at accident conditions (or 38,500 cubic feet per minute at normal operating conditions). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 13).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.11

This SR verifies that the required CRFC unit testing is performed in accordance with the VFTP. The VFTP includes testing HEPA filter performance. The minimum required flow rate through each of the four CRFC units is 33,000 cubic feet per minute at accident conditions (or 38,500 cubic feet per minute at normal operating conditions). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 13).

SR 3.6.6.12 and SR 3.6.6.13

These SRs require verification that each automatic CS valve in the flowpath (860A, 860B, 860C, and 860D) actuates to its correct position and that each CS pump starts upon receipt of an actual or simulated actuation of a containment High pressure signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.14

This SR requires verification that each CRFC unit actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.12 and SR 3.6.6.13, above, for further discussion of the basis for the 24 month Frequency.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.15

This SR requires verification every 24 months that each train of post-accident charcoal filters actuates upon receipt of an actual or simulated safety injection signal. The 24 month frequency is based on engineering judgement and has been shown to be acceptable through operating experience. See SR 3.6.6.12 and SR 3.6.6.13, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.6.16

This SR provides verification that each automatic valve in the NaOH System flow path that is not locked, sealed, or otherwise secured in position (836A and 836B) actuates to its correct position upon receipt of an actual or simulated actuation of a containment Hi-Hi pressure signal. The 24 month frequency is based on engineering judgement and has been shown to be acceptable through operating experience. See SR 3.6.6.12 and SR 3.6.6.13, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.6.17

To ensure that the correct pH level is established in the borated water solution provided by the CS System, flow through the eductor is verified once every 5 years. This SR in conjunction with SR 3.6.6.16 provides assurance that NaOH will be added into the flow path upon CS initiation. A minimum flow of 20 gpm through the eductor must be established as assumed in the accident analyses. A flow path must also be verified from the NaOH tank to the eductors. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow injection.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6.18

With the CS inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 49, 52, 58, 59, 60, and 61, issued for comment July 10, 1967.
 2. Branch Technical Position MTEB 6-1, "pH For Emergency Coolant Water For PWRs."
 3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Automatic Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.
 4. NUREG-0821.
 5. UFSAR, Section 6.3.
 6. UFSAR, Section 6.1.2.4.
 7. 10 CFR 50, Appendix K.
 8. UFSAR, Section 6.2.1.2.
 9. UFSAR, Section 6.2.2.2.
 10. UFSAR, Section 6.5.
 11. UFSAR, Section 6.2.2.1.
 12. ASME, Boiler and Pressure Vessel Code, Section XI.
 13. Regulatory Guide 1.52, Revision 2.
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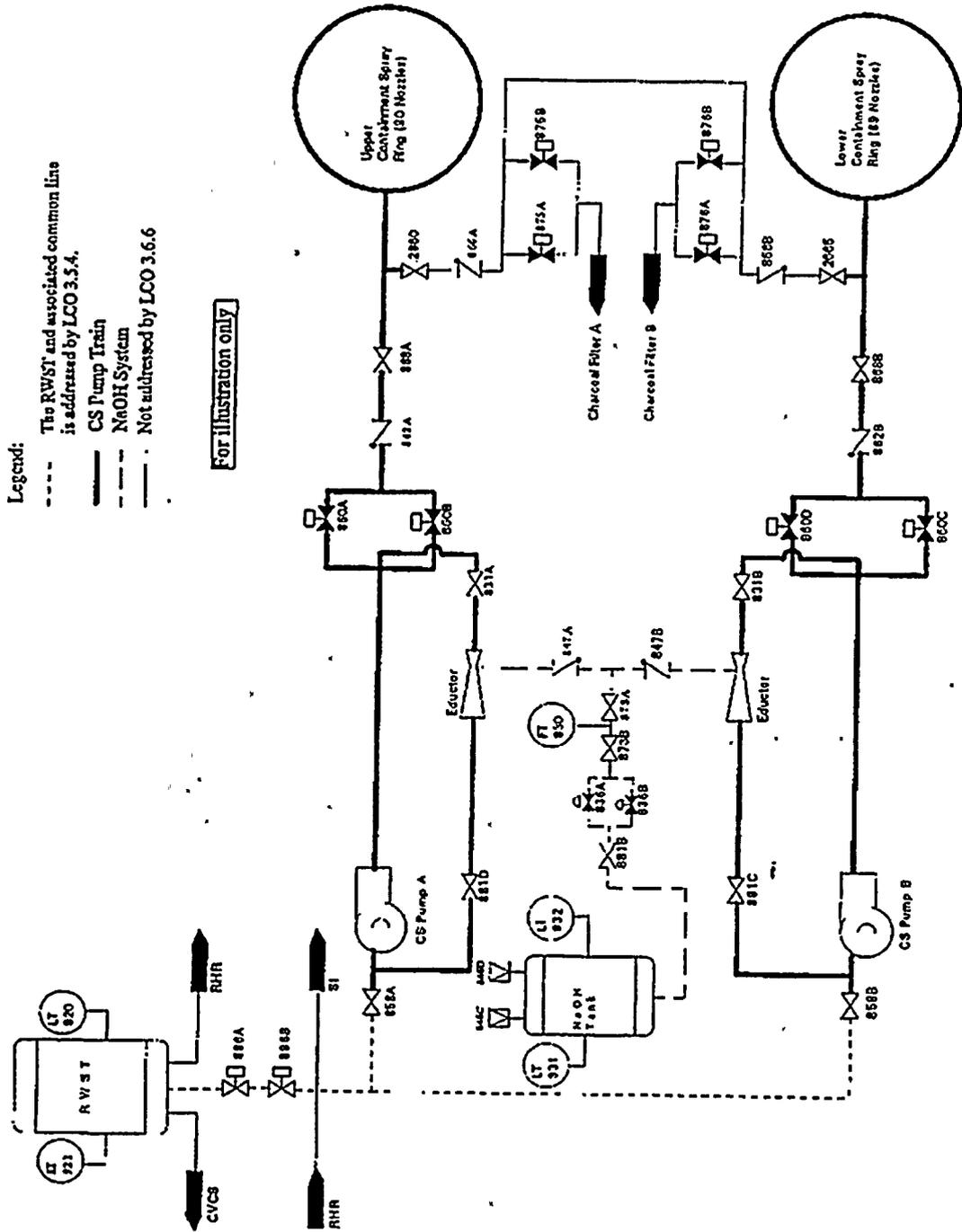
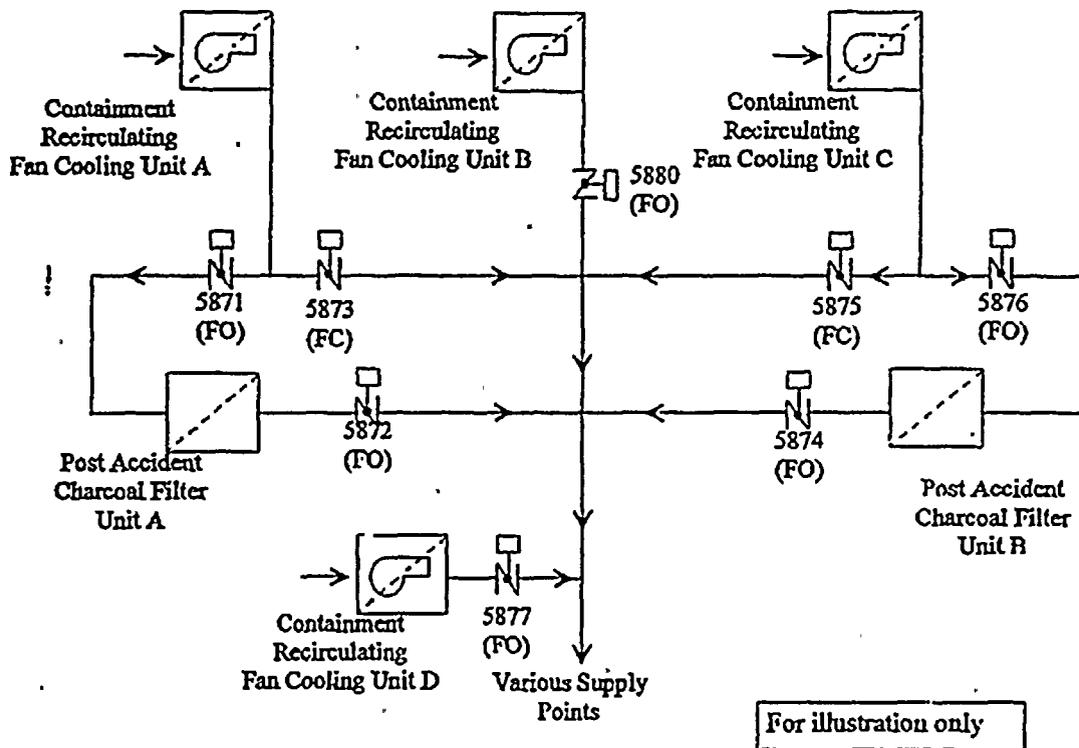


Figure B 3.6.6-1
 Containment Spray and NaOH Systems



Notes:

1. Dampers 5871 and 5872 are associated with Post Accident Charcoal Filter Unit A
2. Dampers 5874 and 5876 are associated with Post Accident Charcoal Filter Unit B
3. Damper 5873 is associated with both CRFC Unit A and Post Accident Charcoal Filter Unit A
4. Damper 5876 is associated with both CRFC Unit C and Post Accident Charcoal Filter Unit B

Figure 2

Figure B 3.6.6-2
 CRFC and Containment Post-Accident Charcoal Systems



B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by collecting the hydrogen and oxygen atmospheric mixture inside containment and oxidizing the hydrogen in a combustion chamber. Additional hydrogen is added by the recombiner to ensure that the noncondensable combustion products that could cause a progressive rise in containment pressure are avoided. Oxygen is also added to prevent depletion of oxygen below the concentration required for stable operation of the combuster. The product of combustion, water vapor, is cooled and condensed from the atmosphere by the Containment Recirculation Fan Cooler System. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA). Prevention of hydrogen accumulation during normal operation is accomplished by use of the Mini-Purge System.

(continued)

BASES

BACKGROUND
(continued)

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the Intermediate Building, a power supply from a separate Engineered Safety Features bus, and a recombiner. The recombiners are comprised of a blower fan to circulate containment air to the combustor, a combustor chamber with a main burner, two igniters (includes an installed spare), a pilot burner, and a dilution chamber downstream of the flame zone where products of the combustion are mixed with containment air to reduce the temperature of the gas leaving the system. A single recombiner is capable of maintaining the hydrogen concentration in containment at approximately 2.0 volume percent (v/o) which is below the 4.1 v/o flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence.

APPLICABLE
SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control prevents a containment wide hydrogen burn, thus ensuring the pressure and temperature inside containment as assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in containment following a LOCA as a result of:

- a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;
- b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;
- c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or
- d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 2 are used to maximize the amount of hydrogen calculated.

The minimum hydrogen flammability limit is 4.1 v/o, however, to avoid a dynamic overpressurization of containment, all hydrogen must be ignited before a concentration of 6.0 v/o is reached (Ref. 3). An alternative to the ignition of hydrogen at concentrations ≥ 6.0 v/o is venting of containment using the Mini-Purge System. However, venting would most likely require evacuations of the general public within a radius of several miles surrounding the plant.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 5.5 v/o about 31 days after the LOCA if no recombiner was functioning (Ref. 3). However, a more realistic model predicts that a hydrogen concentration of 4.1 v/o (the lower flammability limit) will be reached in 31 days. Operation of the hydrogen recombiners ensures that a concentration of 6.0 v/o would not be reached inside containment which could result in an overpressurization given an ignition source.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.1 v/o (Ref. 3).

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Two hydrogen recombiners must be OPERABLE and capable of being placed into operation before the minimum hydrogen flammability limit of 4.1 v/o is reached following a DBA. This ensures operation of at least one hydrogen recombinder in the event of a worst case single active failure. The necessary hydrogen or oxygen required to operate the hydrogen recombinder does not have to be available onsite for the hydrogen recombinder to be considered OPERABLE.

Operation with at least one hydrogen recombinder ensures that the post-LOCA hydrogen concentration can be prevented from exceeding the flammability limit or causing an overpressurization of containment given a hydrogen ignition source.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA or SLB would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a DBA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

(continued)



BASES (continued)

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the Mini-Purge System which consists of two isolation valves per penetration flow path that are capable of opening and a supply fan capable of performing purging functions. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform any Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system (e.g., opening of mini-purge valves). If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.7.1

This SR requires a system functional check of each hydrogen recombiner. A functional check does not require an actual test of the hydrogen recombiner due to the system design which requires oxygen and hydrogen to be pumped into containment. Instead, a functional check is a physical and visual inspection of the hydrogen recombiners to verify that piping is not plugged, the ignitor is OPERABLE, and the recombiners are not fouled. The use of a test gas (e.g., nitrogen) is acceptable. Verification that the recombiners are not fouled requires operation of the blower fan and operation of the system control valves.

The 24 month Frequency for this surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.2

This SR requires performance of a CHANNEL CALIBRATION of each hydrogen recombiner actuation and control channel. A CHANNEL CALIBRATION is required to ensure that the hydrogen recombiner will provide the correct hydrogen/oxygen mixture to the combustion chamber.

The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
 2. Safety Guide 1.7, Rev. 0.
 3. UFSAR, Section 6.2.5.
-

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Eight MSSVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSSVs inoperable.	A.1 Restore inoperable MSSV(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY																		
SR 3.7.1.1	<p>-----NOTE----- Only required to be performed in MODES 1 and 2. -----</p> <p>Verify each MSSV lift setpoint specified below in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 1\%$.</p> <table border="0"> <thead> <tr> <th colspan="2">VALVE NUMBER</th> <th>LIFT SETTING</th> </tr> <tr> <th>SG A</th> <th>SG B</th> <th>(psig +1%, -3%)</th> </tr> </thead> <tbody> <tr> <td>3509</td> <td>3508</td> <td>1140</td> </tr> <tr> <td>3511</td> <td>3510</td> <td>1140</td> </tr> <tr> <td>3515</td> <td>3512</td> <td>1140</td> </tr> <tr> <td>3513</td> <td>3514</td> <td>1085</td> </tr> </tbody> </table>		VALVE NUMBER		LIFT SETTING	SG A	SG B	(psig +1%, -3%)	3509	3508	1140	3511	3510	1140	3515	3512	1140	3513	3514	1085	<p>In accordance with the Inservice Testing Program</p>
VALVE NUMBER		LIFT SETTING																			
SG A	SG B	(psig +1%, -3%)																			
3509	3508	1140																			
3511	3510	1140																			
3515	3512	1140																			
3513	3514	1085																			

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

LCO 3.7.2 Two MSIVs and two non-return check valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 and 3 except when all MSIVs are closed and de-activated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable in flowpath from a steam generator (SG) in MODE 1.	A.1 Restore valve(s) to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. One or more valves inoperable in flowpath from a SG in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed.	8 hours Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and Associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 4.	12 hours
E. One or more valves inoperable in flowpath from each SG.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 Verify closure time of each MSIV is ≤ 5 seconds under no flow and no load conditions.	In accordance with the Inservice Testing Program
SR 3.7.2.2 Verify each main steam non-return check valve can close.	In accordance with the Inservice Testing Program
SR 3.7.2.3 Verify each MSIV can close on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves,
and Main Feedwater Pump Discharge Valves (MFPDVs)

LCO 3.7.3 Two MFRVs, associated bypass valves, and two MFPDVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when both steam generators are isolated from both main feedwater pumps.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFPDV(s) inoperable.	A.1 Close MFPDV(s).	24 hours.
	<u>AND</u> A.2 Verify MFPDV(s) is closed.	Once per 7 days
B. One or more MFRV(s) inoperable.	B.1 Close or isolate MFRV(s).	24 hours
	<u>AND</u> B.2 Verify MFRV(s) is closed or isolated.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One or more MFRV bypass valve(s) inoperable.</p>	<p>C.1 Close or isolate MFRV bypass valve(s).</p> <p><u>AND</u></p> <p>C.2 Verify MFRV bypass valve(s) is closed or isolated.</p>	<p>24 hours</p> <p>Once per 7 days</p>
<p>D. Required Action and associated Completion Time for Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>
<p>E. One or more MFPDV(s) and one or more MFRV(s) inoperable.</p> <p><u>OR</u></p> <p>One or more MFPDV(s) and one or more MFRV bypass valve(s) inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.3.1 Verify the closure time of each MFPDV is \leq 80 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2 Verify the closure time of each MFRV and associated bypass valve is \leq 10 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Two ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System average temperature (T_{avg})
 $\geq 500^{\circ}\text{F}$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARV line inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore ARV line to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3 with $T_{avg} < 500^{\circ}\text{F}$.	8 hours
C. Two ARV lines inoperable.	C.1 Enter LCO 3.0.3.	Immediately



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Perform a complete cycle of each ARV.	24 months
SR 3.7.4.2	Verify one complete cycle of each ARV block valve.	24 months

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Two motor driven AFW (MDAFW) trains, one turbine driven AFW (TDAFW) train, and two standby AFW (SAFW) trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One TDAFW train flowpath inoperable.	A.1 Restore TDAFW train flowpath to OPERABLE status.	7 days
B. One MDAFW train inoperable.	B.1 Restore MDAFW train to OPERABLE status.	7 days
C. TDAFW train inoperable. <u>OR</u> Two MDAFW trains inoperable. <u>OR</u> One TDAFW train flowpath and one MDAFW train inoperable to opposite steam generators (SGs).	C.1 Restore one MDAFW train, or TDAFW train flowpath to OPERABLE status.	72 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. All AFW trains to one or more SGs inoperable.	D.1 Restore one AFW train or TDAFW flowpath to each affected SG to OPERABLE status.	4 hours
E. One SAFW train inoperable.	E.1 Restore SAFW train to OPERABLE status.	14 days
F. Both SAFW trains inoperable.	F.1 Restore one SAFW train to OPERABLE status.	7 days
G. Required Action and associated Completion Time for Condition A, B, C, D, E, or F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	6 hours 12 hours
H. Three AFW trains and both SAFW trains inoperable.	H.1 -----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status. ----- Initiate action to restore one MDAFW, TDAFW, or SAFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.1 Verify each AFW and SAFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.5.2 -----NOTE----- Required to be met prior to entering MODE 1 for the TDAFW pump. ----- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.3 Verify the developed head of each SAFW pump at the flow test point is greater than or equal to the required developed head.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.4 Perform a complete cycle of each AFW and SAFW motor operated suction valve from the Service Water System, each AFW and SAFW discharge motor operated isolation valve, and each SAFW cross-tie motor operated valve.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.5 Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.6 -----NOTE----- Required to be met prior to entering MODE 1 for the TDAFW pump. -----</p> <p>Verify each AFW pump starts automatically on an actual or simulated actuation signal.</p>	<p>24 months</p>
<p>SR 3.7.5.7 Verify each SAFW train can be actuated and controlled from the control room.</p>	<p>24 months</p>



3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LC0 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST water volume not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours
	<u>AND</u> A.2 Restore CST water volume to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify the CST water volume is $\geq 22,500$ gal.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains, two CCW heat exchangers, and the CCW loop header shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW train inoperable.	A.1 Restore CCW train to OPERABLE status.	72 hours
B. One CCW heat exchanger inoperable.	B.1 Restore CCW heat exchanger to OPERABLE status.	31 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Two CCW trains, two CCW heat exchangers, or loop header inoperable.</p>	<p>-----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status. -----</p>	
	<p>D.1 Initiate Action to restore one CCW train, one heat exchanger, and loop header to OPERABLE status.</p>	<p>Immediately</p>
	<p><u>AND</u></p>	
	<p>D.2 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u></p>	
	<p>D.3 Be in MODE 4.</p>	<p>12 hours</p>



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.7.1 -----NOTE----- Isolation of CCW flow to individual components does not render the CCW loop header inoperable. -----</p> <p>Verify each CCW manual and power operated valve in the CCW train and heat exchanger flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.7.2 Perform a complete cycle of each motor operated isolation valve to the residual heat removal heat exchangers.</p>	<p>In accordance with the Inservice Testing Program</p>

3.7 PLANT SYSTEMS

3.7.8 Service Water (SW) System

LCO 3.7.8 Two SW trains and the SW loop header shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW train inoperable.	A.1 Restore SW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours
C. Two SW trains or loop header inoperable.	C.1 -----NOTE----- Enter applicable conditions and Required Actions of LCO 3.7.7, "CCW System," for the component cooling water heat exchanger(s) made inoperable by SW. ----- Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify screenhouse bay water level and temperature are within limits.	24 hours
SR 3.7.8.2	<p>-----NOTE----- Isolation of SW flow to individual components does not render the SW loop header inoperable. -----</p> <p>Verify each SW manual, power operated, and automatic valve in the SW train flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	31 days
SR 3.7.8.3	Verify all SW loop header cross-tie valves are locked in the correct position.	31 days
SR 3.7.8.4	Verify each SW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.8.5	Verify each SW pump starts automatically on an actual or simulated actuation signal.	24 months



3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Air Treatment System (CREATS)

LCO 3.7.9 The CREATS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. CREATS filtration train inoperable.</p>	<p>A.1 Restore CREATS filtration train to OPERABLE status.</p> <p><u>OR</u></p> <p>A.2 -----NOTE----- The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition. ----- Place isolation dampers in CREATS Mode F.</p>	<p>48 hours</p> <p>48 hours</p>
<p>B. -----NOTE----- Separate Condition entry allowed for each damper. ----- One CREATS isolation damper in one or more outside air flowpaths inoperable.</p>	<p>B.1 Restore isolation damper to OPERABLE status.</p>	<p>7 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.</p>	<p>C.1 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u> C.2 Be in MODE 5.</p>	<p>36 hours</p>
<p>D. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6 or during movement of irradiated fuel.</p>	<p>D.1 Place OPERABLE isolation damper(s) in CREATS Mode F.</p>	<p>Immediately</p>
	<p><u>OR</u> D.2.1 Suspend CORE ALTERATIONS.</p>	<p>Immediately</p>
	<p><u>AND</u> D.2.2 Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately</p>
<p>E. Two CREATS isolation dampers for one or more outside air flow paths inoperable in MODE 1, 2, 3, or 4.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREATS isolation dampers for one or more outside air flow paths inoperable in MODE 5 or 6 or during movement of irradiated fuel assemblies.	F.1 Initiate actions to restore one isolation damper to OPERABLE status.	Immediately
	<u>AND</u> F.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> F.3 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate the CREATS filtration train \geq 15 minutes.	31 days
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.9.3	Verify the CREATS actuates on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.10 Auxiliary Building Ventilation System (ABVS)

LCO 3.7.10 The ABVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the Auxiliary Building when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ABVS inoperable.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the Auxiliary Building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.10.1 Verify ABVS is in operation.	24 hours
SR 3.7.10.2 Verify ABVS maintains a negative pressure with respect to the outside environment at the Auxiliary Building operating floor level.	24 hours

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.7.10.3 Perform required Spent Fuel Pool Charcoal Adsorber System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool (SFP) Water Level

LCO 3.7.11 The SFP water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify the SFP water level is \geq 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.12 The SFP boron concentration shall be ≥ 300 ppm.

APPLICABILITY: When fuel assemblies are stored in the SFP and a SFP verification has not been performed since the last movement of fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. SFP boron concentration not within limit.</p>	<p>-----NOTE----- LCO 3.0.3 is not applicable. -----</p>	
	<p>A.1 Suspend movement of fuel assemblies in the SFP.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>A.2.1 Initiate action to restore SFP boron concentration to within limit.</p>	<p>Immediately</p>
	<p><u>OR</u></p> <p>A.2.2 Initiate action to perform SFP verification.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.12.1 Verify the SFP pool boron concentration is within limit.	31 days

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Storage

LCO 3.7.13 Fuel assembly storage in the spent fuel pool shall be maintained as follows:

- a. Fuel assemblies in Region 1 shall have a K-infinity of ≤ 1.458 ; and
- b. Fuel assemblies in Region 2 shall have initial enrichment and burnup within the acceptable area of the Figure 3.7.13-1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Requirements of the LCO not met for either region.</p>	<p>A.1 -----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>Initiate action to move the noncomplying fuel assembly from the applicable region.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.13.1 -----NOTE----- Not required to be performed when transferring a fuel assembly from Region 2 to Region 1. ----- Verify by administrative means the K-infinity of the fuel assembly is ≤ 1.458.</p>	<p>Prior to storing the fuel assembly in Region 1</p>
<p>SR 3.7.13.2 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-1.</p>	<p>Prior to storing the fuel assembly in Region 2</p>

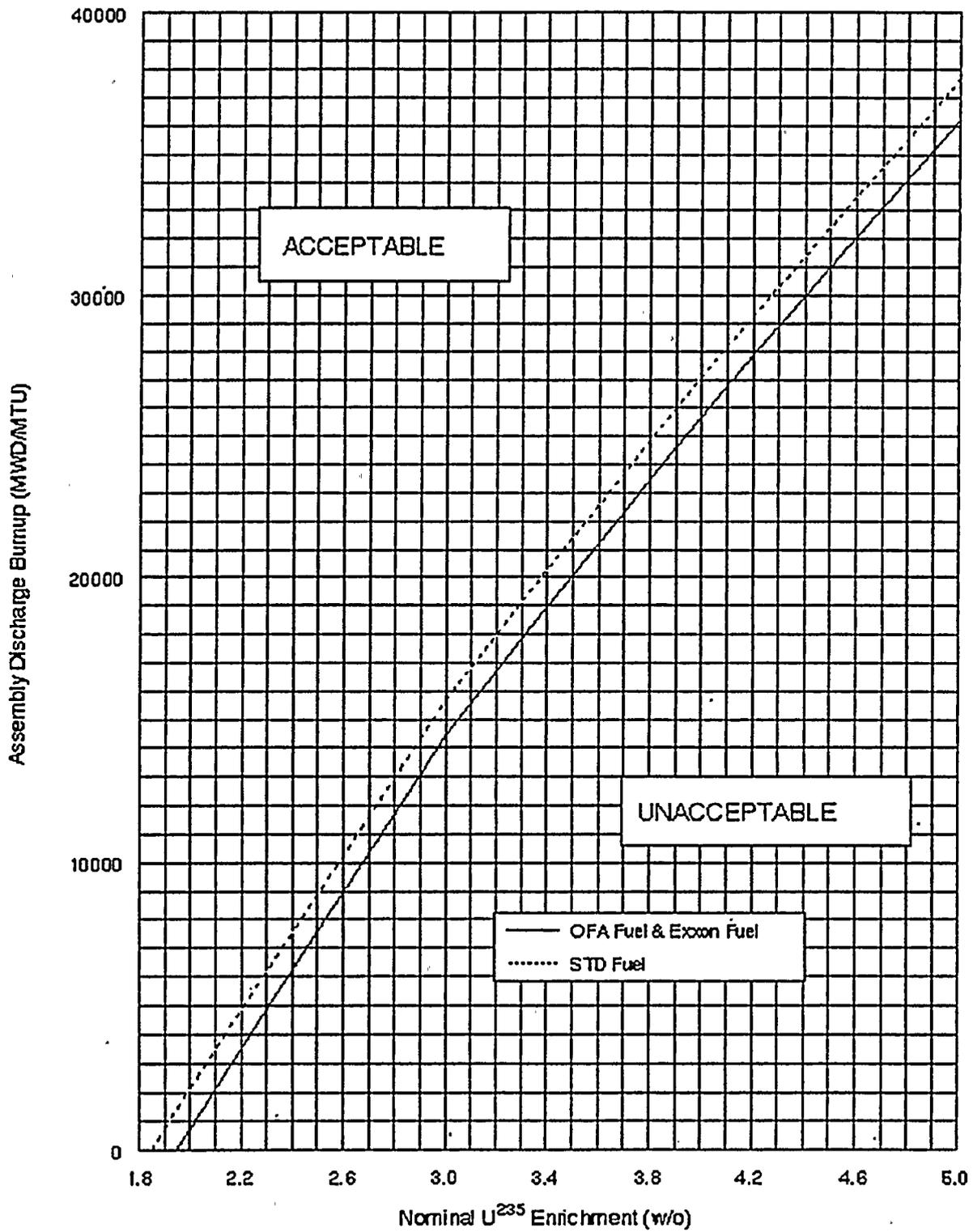


Figure 3.7.13-1
Fuel Assembly Burnup Limits in Region 2

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	<u>AND</u> A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the specific activity of the secondary coolant is $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	31 days

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred (but non safety related) heat sink, provided by the condenser and circulating water system, is not available.

Four MSSVs are located on each main steam header, outside containment in the Intermediate Building, upstream of the main steam isolation valves (Ref. 1). MSSVs 3509, 3511, 3513, and 3515 are located on the steam generator (SG) A main steam header while MSSVs 3508, 3510, 3512 and 3514 are located on the SG B main steam header. The MSSVs are designed to limit the secondary system to $\leq 110\%$ of design pressure when passing 100% of design flow. The MSSV design includes staggered setpoints so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine/reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs is to limit the secondary system pressure to $\leq 110\%$ of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (A00) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased RCS heat removal events (Ref. 2). Of these, the full power loss of external load event is the limiting A00. This event also results in the loss of normal feedwater flow to the SGs.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The transient response for a loss of external load event without a direct reactor trip (i.e., loss of load when < 50% RTP) presents no hazard to the integrity of the RCS or the Main Steam System. For transients at power levels > 50%, the effect on RCS safety limits is evaluated with no credit taken for the pressure relieving capability of pressurizer spray, the steam dump system, and the SG atmospheric relief valves. The reactor is tripped on high pressurizer pressure with the pressurizer safety valves and MSSVs required to be opened to maintain the RCS and Main Steam System within 110% of their design values.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening (as an initiating event only), and failure to reclose once opened. The passive failure mode is failure to open upon demand which is not considered in the accident analyses.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve SG overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to SR 3.7.1.1 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or secondary system.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, four MSSVs per SG are required to be OPERABLE to ensure that the RCS remains within its pressure safety limit and that the secondary system, from the SGs to the main steam isolation valves, is limited to $\leq 110\%$ of design pressure for all DBAs.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The SGs are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, the assumptions used in the accident analysis for loss of external load may no longer be valid and the safety valve(s) must be restored to OPERABLE status within 4 hours. This Condition specifically addresses the appropriate ACTIONS to be taken in the event that a non-significant discrepancy related to the MSSVs is discovered with the plant operating in MODES 1, 2, or 3. Examples of this type of discrepancy include administrative (e.g., documentation of inspection results) or similar deviations which do not result in a loss of MSSV capability to relieve steam. The 4 hour Completion Time allows a reasonable period of time for correction of administrative only problems or for the plant to contact the NRC to discuss appropriate action. The 4 hour Completion time is based on engineering judgement.

This Condition is not applicable to a situation in which the ability of a MSSV to open or reclose is questionable. In this event, this Condition is no longer applicable and Condition B of this LCO should be entered immediately since no corrective actions can be implemented during MODES 1, 2, and 3.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the MSSV(s) cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 3), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 4). According to Reference 4, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Setpoint pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria; and
- e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. This SR allows a +1% and -3% setpoint tolerance for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.2.4.
 2. UFSAR, Section 15.2.
 3. ASME, Boiler and Pressure Vessel Code, Section XI.
 4. ANSI/ASME OM-1-1987.
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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

BASES

BACKGROUND

The MSIVs (3516 and 3517) isolate steam flow from the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). MSIV closure is necessary to isolate a SG affected by a steam generator tube rupture (SGTR) event or a steam line break (SLB) to stop the loss of SG inventory and to protect the integrity of the unaffected SG for decay heat removal. The MSIVs are air operated swing disk check valves that are held open by an air operator against spring pressure. The MSIVs are installed to use steam flow to assist in the closure of the valve (Ref. 1).

A MSIV is located in each main steam line header outside containment in the Intermediate Building. The MSIVs are downstream from the main steam safety valves (MSSVs) and turbine driven auxiliary feedwater (AFW) pump steam supply, to assure the MSSVs prevent overpressure on the secondary side and assure steam is available to the AFW system following MSIV closure. Closing the MSIVs isolates each SG from the other, and isolates the turbine, steam dump system, and other auxiliary steam supplies from the SGs.

The MSIVs close on a main steam isolation signal generated by either high containment pressure, high steam flow coincident with low T_{avg} and safety injection (SI), or high-high steam flow coincident with SI.

The MSIVs are designed to work with non-return check valves (3518 and 3519) located immediately downstream of each MSIV to preclude the blowdown of more than one SG following a SLB. The MSIVs fail closed on loss of control or actuation power and loss of instrument air once the air is bled off from the supply line. The MSIVs may also be actuated manually.

Each MSIV has a normally closed manual MSIV bypass valve.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The design basis of the MSIVs and non-return check valves is established by the large SLB (Ref. 2). The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity. The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MSIV on the intact SG for this case is assumed to close to prevent excessive cooldown of the RCS which could result in a lower DNB ratio.

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. Due to the non-return check valve on the faulted SG, reverse flow from the steam headers downstream of the MSIV and from the intact SG is prevented from contributing to the energy and mass released inside containment by the SLB. This check valve is a passive device which is not assumed to fail.

SLBs outside of containment can occur in the Intermediate Building and downstream of the MSIVs in the Turbine Building. A SLB in piping > 6 inches diameter in the Intermediate Building is not required to be considered due to an augmented piping inspection program (Ref. 3). For a SLB in the Turbine building, the MSIVs on both SGs must close to isolate the break and terminate the event.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The MSIVs are also credited in a SGTR to manually isolate the SG with the ruptured tube. In addition to minimizing the radiological releases, this assists the operator in isolating the RCS flow through the ruptured SG by preventing the SG from continuing to depressurize and creating a higher pressure difference between the secondary system and the primary system.

The MSIVs are also considered in other DBAs such as the feedwater line break in which closure of the MSIV on the intact SG maximizes the effect of the break since the energy removal capability of the intact SG would be reduced with respect to long term heat removal.

In addition to providing isolation of a faulted SG during a SLB, feedwater line break, or a SGTR, the MSIVs also serve as a containment isolation boundary. The MSIVs are the second containment isolation boundary for the main steam line penetrations which use the steam lines and SGs inside containment as the first boundary. The MSIVs do not receive an automatic containment isolation signal since a spurious signal could result in a significant plant transient.

The MSIVs and non-return check valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that two MSIVs and the non-return check valves in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits and they can close on an isolation actuation signal. A MSIV must also be capable of isolating a SG for containment isolation purposes. The non-return check valves are considered OPERABLE when they are capable of closing.

This LCO provides assurance that the MSIVs and non-return check valves will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

(continued)

BASES (continued)

APPLICABILITY The MSIVs and non-return check valves must be OPERABLE in MODES 1, 2, and 3 when there is significant mass and energy in the RCS and SGs to challenge the integrity of containment, or allow a transient to approach DNBR limits. When the MSIVs are closed and de-activated in MODES 2 and 3, they are already performing their safety function and the MSIVs and their associated non-return check valves are not required to be OPERABLE per this LCO.

In MODE 4, the MSIVs and non-return check valves are normally closed, and the RCS and SG energy is low.

In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and non-return check valves are not required for isolation of potential main steam pipe breaks in these MODES.

ACTIONS The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one or more valves inoperable in flow path from a SG in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to these valves can be made with the plant under hot conditions. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and non-return check valves and the ability to isolate the affected SG by turbine stop valves.

The 8 hour Completion Time is greater than that normally allowed for containment isolation boundaries because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from most other containment isolation boundaries in that the closed system provides an additional means for containment isolation. Failure of this closed system can only result from a SGTR which is not postulated to occur with any other DBA (e.g., LOCA).

(continued)

BASES

ACTIONS
(continued)

B.1

If the MSIV and/or non-return check valve from a SG cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner without challenging plant systems.

C.1 and C.2

Since the MSIVs and non-return check valve are required to be OPERABLE in MODES 2 and 3, the inoperable valve(s) may either be restored to OPERABLE status or the associated MSIV closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis and the non-return check valve is no longer required.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable valves that cannot be restored to OPERABLE status within the specified Completion Time, but the associated MSIV is closed, the MSIV must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgement, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

(continued)



BASES

ACTIONS
(continued)

D.1 and D.2

If the MSIVs and/or non-return check valve cannot be restored to OPERABLE status or the associated MSIV is not closed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 conditions in an orderly manner without challenging plant systems.

E.1

If one or more valves in the flow path from each SG are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MSIVs and non-return check valves are inoperable such that at least one valve is inoperable in each of the two main steam flow paths.

SURVEILLANCE
REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds under no flow and no load conditions. The MSIVs are swing-disk check valves that are held open by their air operators against spring pressure. Once the MSIVs begin to close during hot conditions, the steam flow will assist the valve closure such that testing under no flow and no load conditions is conservative. The 5 second closure time is consistent with the expected response time for instrumentation associated with the MSIV and the accident analysis assumptions.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.2.2

This SR verifies that each main steam non-return check valve can close. As the non-return check valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.

SR 3.7.2.3

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MSIVs should not be tested at power, since even a partial stroke exercise increases the risk of a valve closure and plant transient when the plant is above MODE 4. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODES 1, 2 and 3.

The frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.4.4.
 2. UFSAR, Section 15.1.5.
 3. UFSAR, Section 3.6.2.5.1.
 4. 10 CFR 100.11.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
-

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves,
and Main Feedwater Pump Discharge Valves (MFPDVs)

BASES

BACKGROUND

The MFRVs (4269 and 4270) and their associated bypass valves (4271 and 4272), and MFPDVs (3977 and 3976) isolate main feedwater (MFW) flow to the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). The safety related function of the MFRVs, associated bypass valves, and MFPDVs is to provide for isolation of MFW flow to the secondary side of the SGs terminating the DBA for line breaks occurring downstream of the valves. Closure effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release for steam line breaks (SLBs) or feedwater line breaks (FWLBs) inside containment, and reducing the cooldown effects for SLBs.

The MFRVs, associated bypass valves, and MFPDVs in conjunction with check valves located downstream of the isolation valves also provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact SG (see Figure B 3.7.3-1).

One MFPDV is located in the Turbine Building on the discharge line of each MFW pump (Ref. 1). One MFRV and associated bypass valve is located on each MFW line to its respective SG, outside containment in the Turbine Building. The MFRVs, associated bypass valves, and MFPVs are located upstream of the AFW injection point so that AFW may be supplied to the SGs following closure of the MFRVs and bypass valves. The piping volume from these valves to the SGs is accounted for in calculating mass and energy releases, and must be refilled prior to AFW reaching the SG following either an SLB or FWLB.

(continued)

BASES

BACKGROUND
(continued)

The MFPDV closes on the opening of the MFW pump breaker which occurs on receipt of a safety injection signal or any other signal which trips the pump breaker. The MFRVs and bypass valves close on receipt of a safety injection signal, a SG high level signal, or on a reactor trip with $T_{avg} < 554^{\circ}\text{F}$ with the associated MFRV in auto. All valves may also be actuated manually. In addition to the MFRVs, associated bypass valves and MFPDVs, a check valve located outside containment for each feedwater line is available. The check valve isolates the feedwater line penetrating containment providing a containment isolation boundary.

APPLICABLE
SAFETY ANALYSES

The design basis of the MFRVs, associated bypass valves, and MFPDVs is established by the analyses for the SLB. The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity (Ref. 2). The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MFRV and bypass valve on the intact SG for this case are assumed to close on a safety injection signal to prevent excessive cooldown of the RCS which could result in a lower DNB ratio. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps, which have their breakers opened by a SI signal, and the MFPDV which close on opening of the MFW pump breakers.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant, maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. The MFRV and bypass valve on the faulted SG are assumed to close on a safety injection signal to prevent continued contribution to the energy and mass released inside containment by the SLB. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps and closure of the MFPDVs.

The MFRVs and bypass valves are also credited for isolation in the feedwater transient analyses (e.g., increase in feedwater flow). These valves close on either a safety injection or high SG level signal depending on the scenario. The valves also must close on a FWLB to limit the amount of additional mass and energy delivered to the SGs and containment.

The failure of the MFRVs to control flow is also considered as an initiating event. This includes consideration of a valve failure coincident with an atmospheric relief valve failure since a single component in the Advanced Digital Feedwater Control System (ADFCS) controls both components (Ref. 3). This combined valve failure accident scenario is evaluated with respect to DNB since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the SLB accident.

The MFRVs, associated bypass valves, and MFPDVs satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO This LCO ensures that the MFRVs, associated bypass valves, and MFPDVs will isolate MFW flow to the SGs, following a FWLB or SLB.

This LCO requires that two MFPDVs, two MFRVs, and two MFRV bypass valves be OPERABLE. The MFRVs, associated bypass valves, and MFPDVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. It may also result in the introduction of water into the main steam lines for an excess feedwater flow event.

APPLICABILITY

The MFRVs, associated bypass, and MFPDVs valves must be OPERABLE whenever there is significant mass and energy in the RCS and SGs. This ensures that, in the event of a DBA, the accident analysis assumptions are maintained. In MODES 1, 2, and 3, the MFRVs, associated bypass valves, and MFPDVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve such that both SGs are isolated from both MFW pumps, they are already performing their safety function and no longer required to be OPERABLE.

In MODE 4, the MFRVs, associated bypass valves, and MFPDVs are normally closed since AFW is providing decay heat removal due to the low SG energy level. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MFRVs, associated bypass valves, and MFPDVs are not required for isolation of potential pipe breaks in these MODES.

(continued)

BASES (continued)

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one or more MFPDV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or close the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFPDV that is closed must be verified on a periodic basis that it remains closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

B.1 and B.2

With one or more MFRV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV that is closed must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

With one or more MFRV bypass valve(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV bypass valve that is closed must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

D.1 and D.2

If the MFRV, associated bypass valve, or MFPDV cannot be restored to OPERABLE status or closed within 24 hours or cannot be verified closed once per 31 days, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

E.1

If one or more MFPDV(s) and one or more MFRV(s), or one or more MFPDV(s) and one or more MFRV bypass valve(s) are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MFRVs, associated bypass valves, or MFPDVs are inoperable such that a MFW pump, condensate pump, or condensate booster pump can provide unisolable flow to one or both SGs (see Figure B 3.7.3-1).

SURVEILLANCE
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFPDV is ≤ 80 seconds from the full open position on an actual or simulated actuation signal (i.e., from opening of MFW pump breakers). The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI, (Ref. 4) requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.3.2

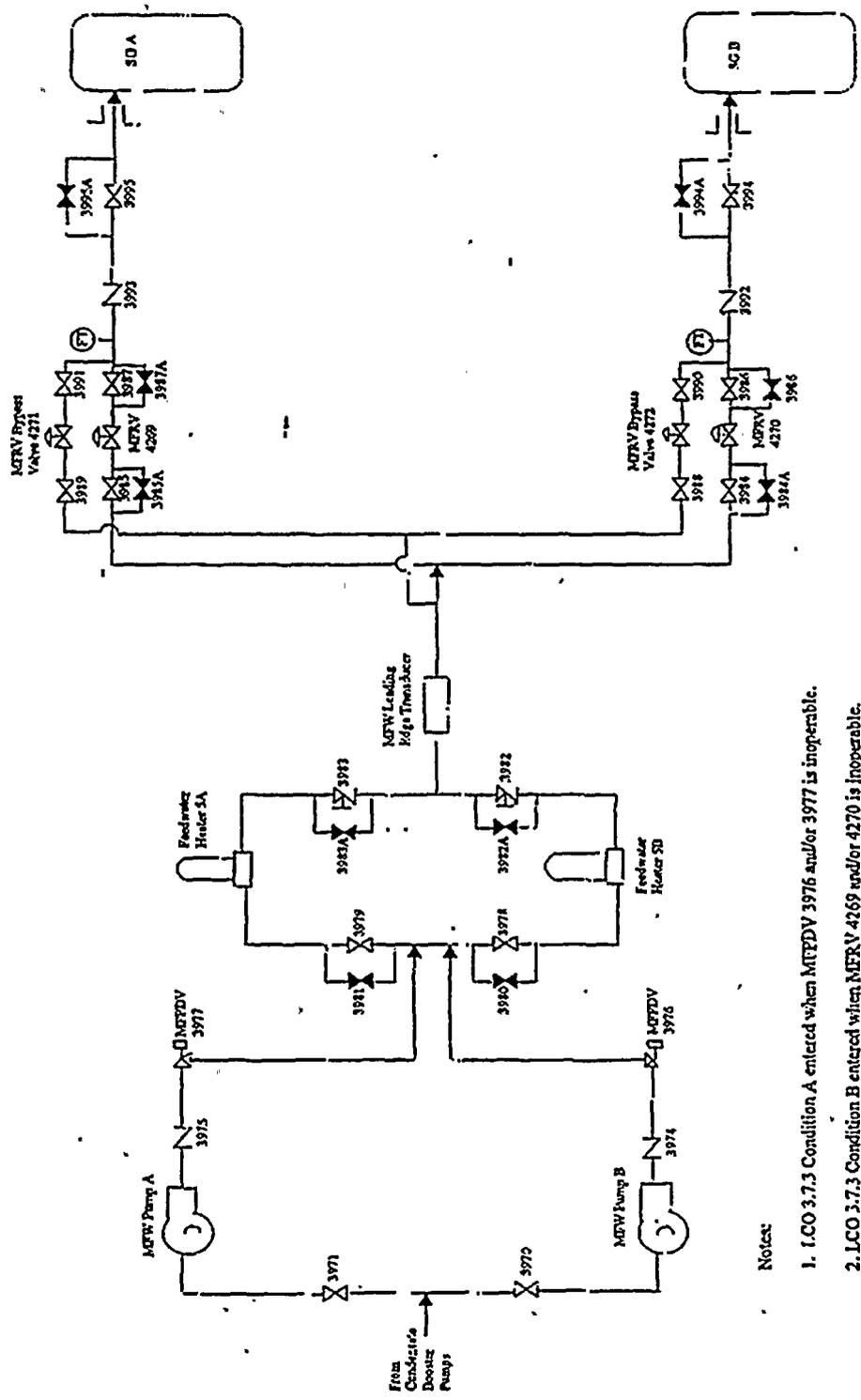
This SR verifies that the closure time of each MFRV and associated bypass valve is ≤ 10 seconds from the full open position on an actual or simulated actuation signal. The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 4), requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 10.4.5.3.
 2. UFSAR, Section 15.1.5.
 3. UFSAR, Section 15.1.6.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
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MFRVs, Associated Bypass Valves, and MFPDVs
B 3.7.3



For illustration only

Notes:

1. LCO 3.7.3 Condition A entered when MFPDV 3976 and/or 3977 is inoperable.
2. LCO 3.7.3 Condition B entered when MFRV 4269 and/or 4270 is inoperable.
3. LCO 3.7.3 Condition C entered when MFRV Bypass Valve 4271 and/or 4272 is inoperable.
4. LCO 3.7.3 Condition D entered when any combination of valve inoperabilities results in an unsatisfiable flowpath from the condensate booster pumps to one or more SGs.

Figure B 3.7.3-1
MFRVs, Associated Bypass Valves and MFPDVs

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Relief Valves (ARVs)

BASES

BACKGROUND

There is an ARV (3410 and 3411) located on the main steam header from each steam generator (SG). The ARVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ARVs have two functions (Ref. 1):

- a. provide secondary system overpressure protection below the setpoint of the main steam safety valves (MSSVs); and
- b. provide a method for cooling the plant should the preferred heat sink via the steam dump system to the condenser not be available.

The accident analyses do not credit either of these functions since the ARVs do not have a safety-related source of motive air and the accident analyses do not typically require cooldown to the residual heat removal entry conditions since the plant was originally designed to maintain Hot Shutdown conditions indefinitely. The only exception is with respect to steam generator tube rupture (SGTR) events which require the use of at least one ARV to provide heat removal from the Reactor Coolant System (RCS) to prevent saturation conditions from developing.

The ARVs are air operated valves located in the Intermediate Building with a relief capacity of 329,000 lbm/hr each (approximately 5% of RTP). The ARVs are normally closed, fail closed valves which receive motive air from the instrument air system. The valves can also receive motive air from a non-seismic backup nitrogen bottle bank system. The valves are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are normally controlled by the Advanced Digital Feedwater Control System (ADFCS) but can also be remote manually operated and opened locally by use of handwheels located on the valves.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The design basis for the ARVs is established by the SGTR event (Ref. 2). For this accident scenario, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. Following a SGTR, the MSSVs will maintain the secondary system pressure at approximately 1085 psig which could result in the loss of subcooling margin since the RCS average temperature is attempting to stabilize at approximately 547°F. The ARVs are used during the first 30 to 60 minutes of the SGTR to continue the RCS cooldown in an effort to reduce, and eventually terminate, the primary to secondary system flow in the ruptured SG. The inability to cooldown could result in inadequate subcooling margin which would delay the termination of the leakage through the ruptured tube.

The opening of the ARVs is also considered coincident with a failure of a main feedwater regulating valve (Ref. 3) since a single component in the ADFCS controls both components. This combined valve failure accident scenario is evaluated with respect to departure from nucleate boiling since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the steam line break accident.

The ARVs are equipped with block valves in the event the ARV spuriously fails to open or fails to close during use.

The ARVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two ARVs and their associated manual block valves are required to be OPERABLE. The ARVs are required for manual operation either locally (using the handwheel or local panel) or remotely to relieve main steam pressure. The ARV block valves must be OPERABLE to isolate a failed open ARV. A closed block valve does not render it or its ARV line inoperable if operator action time to open the block valve can be accomplished within the time frames specified below. Failure to meet the LCO can result in the inability to cool the plant following a SGTR event in which the condenser is unavailable for use with the steam dump system.

(continued)

BASES

LCO
(continued)

An ARV line is considered OPERABLE when it is capable of being manually opened within 20 minutes of determining the need to utilize the ARV following a SGTR. The ARV line must also be capable of closing within 15 minutes in the event the ARV spuriously opens on the SG with the ruptured tube. Finally, the ARV line must be capable of closing within 5 minutes in the event that the ARV on the intact SG fails to close following initiation of a cooldown. For the closure requirements, either the ARV or its associated block valve may be credited for OPERABILITY.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, the ARV lines are required to be OPERABLE.

In MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODE 4, the ARVs are not required since the saturation pressure of the reactor coolant is below the lift settings of the MSSVs. In MODE 5 or 6, an SGTR is not a credible event since the water in the SGs is below the boiling point and RCS pressure is low.

ACTIONS

A.1

With one ARV line inoperable, action must be taken to restore the valve to OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ARV line and a nonsafety grade backup in the steam dump system.

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply since the steam dump system would normally be in service during lower MODES of operation and can provide an acceptable alternative to the inoperable ARV line.

(continued)

BASES

ACTIONS
(continued)

B.1

If the ARV line cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 with RCS average temperature < 500°F within 8 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both ARV lines are inoperable, the plant is in a condition outside of the accident analyses for a SGTR event; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.7.4.1

To perform a cooldown of the RCS, the ARVs must be able to be opened either remotely or locally. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the block valve is to isolate a failed open ARV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 10.3.2.5.
 2. UFSAR, Section 15.6.3.
 3. UFSAR, Section 15.1.6.
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B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System supplies feedwater to the steam generators (SGs) to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the main steam safety valves (MSSVs) or atmospheric relief valves. If the main condenser is available, steam may be released via the steam dump valves. The AFW System is comprised of two separate systems, a preferred AFW System and a Standby AFW (SAFW) System (Ref. 1).

AFW System

The preferred AFW System consists of two motor driven AFW (MDAFW) pumps and one turbine driven AFW (TDAFW) pump configured into three separate trains which are all located in the Intermediate Building (see Figure B 3.7.5-1). Each MDAFW train provides 100% of AFW flow capacity, and the TDAFW pump provides 200% of the required capacity to the SGs, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to the condensate storage tanks (CSTs). Each MDAFW train is powered from an independent Class 1E power supply and feeds one SG, although each pump has the capability to be realigned from the control room to feed the other SG via cross-tie lines containing normally closed motor operated valves (4000A and 4000B). The two MDAFW trains will actuate automatically on a low-low level signal in either SG, opening of the main feedwater (MFW) pump breakers, a safety injection (SI) signal, or the ATWS mitigation system actuation circuitry (AMSAC). The pumps can also be manually started from the control room.

(continued)

BASES

BACKGROUND
(continued)

The TDAFW pump receives steam from each main steam line upstream of the two main steam isolation valves. Either of the steam lines will supply 100% of the requirements of the TDAFW pump. The TDAFW pump supplies a common header capable of feeding both SGs by use of fail-open, air-operated control valves (4297 and 4298). The TDAFW pump will actuate automatically on a low-low level signal in both SGs, loss of voltage on 4160 V Buses 11A and 11B, or the ATWS mitigation system actuation circuitry (AMSAC). The pump can also be manually started from the control room.

The normal source of water for the AFW System is the CSTs which are located in the non-seismic Service Building. The Service Water (SW) System (LCO 3.7.8) can also be used to supply a safety-related source of water through normally closed motor operated valves (4013, 4027, and 4028) which supply each AFW train.

SAFW System

The SAFW System consists of two motor driven pumps configured into two separate trains (see Figure B 3.7.5-2). Each motor driven SAFW train provides 100% of the AFW flow capacity as assumed in the accident analyses and supplies one SG through the use of a normally open motor-operated stop check valve. Each pump has the capability to be realigned from the control room to feed the other SG via normally closed motor operated valves (9703A and 9703B). Each pump is powered from an independent Class 1E power supply and can be powered from the diesel generators provided that the breaker for the associated MDAFW pump is opened. The safety-related source of water for the SAFW System is the SW System through two normally closed motor operated valves (9629A and 9629B). Condensate can also be supplied by a 10,000 gallon condensate test tank and the yard fire hydrant yard loop.

The SAFW System is manually actuated in the event that the preferred AFW System has failed due to a high energy line break (HELB) in the Intermediate Building, a seismic or fire event. The SAFW trains are located in the SAFW Pump Building located adjacent to the Auxiliary Building.

(continued)

BASES

BACKGROUND
(continued)

The SAFW Pump Building environment is controlled by room coolers which are supplied by the same SW header as the pump trains. These coolers are required when the outside air temperature is $\geq 80^{\circ}\text{F}$ to ensure the SAFW Pump Building remains $\leq 120^{\circ}\text{F}$ during accident conditions.

The AFW System is designed to supply sufficient water to the SG(s) to remove decay heat with SG pressure at the lowest MSSV set pressure plus 1%. Subsequently, the AFW System supplies sufficient water to cool the plant to RHR entry conditions, with steam released through the ARVs.

APPLICABLE
SAFETY ANALYSES

The design basis of the AFW System is to supply water to the SG(s) to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the SGs at pressures corresponding to the lowest MSSV set pressure plus 1%.

The AFW System mitigates the consequences of any event with the loss of normal feedwater. The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows (Ref. 2):

- a. Feedwater Line Break (FWLB);
- b. Loss of MFW (with and without offsite power);
- c. Steam Line Break (SLB);
- d. Small break loss of coolant accident (LOCA);
- e. Steam generator tube rupture (SGTR); and
- f. External events (tornados and seismic events).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The AFW System design is such that any of the above DBAs can be mitigated using the preferred AFW System or SAFW System. For the FWLB, SLB, and external events DBAs (items a, c, and f), the worst case scenario is the loss of all three preferred AFW trains due to a HELB in the Intermediate or Turbine Building, or a failure of the Intermediate Building block walls. For these three events, the use of the SAFW System within 10 minutes is assumed by the accident analyses. Since a single failure must also be assumed in addition to the HELB or external event, the capability of the SAFW System to supply flow to an intact SG could be compromised if the SAFW cross-tie is not available. For HELBs within containment, use of either the SAFW System or the AFW System to the intact SG is assumed within 10 minutes.

For the SGTR events (item e), the accident analyses assume that one AFW train is available upon a SI signal or low-low SG level signal. Additional inventory is being added to the ruptured SG as a result of the SGTR such that AFW flow is not a critical feature for this DBA.

For the loss of MFW events and small break LOCA (items b and d), two trains of AFW are assumed available (i.e., two MDAFW trains or the TDAFW train) upon a low-low SG level signal and SI signal, respectively. Two AFW trains are assumed available since no single failure can result in the loss of more than one AFW train. The loss of MFW is a Condition 2 event (Ref. 3) which places limits on the response of the RCS from the transient (e.g., no challenge to the pressurizer power operated relief valves is allowed). Two trains of AFW are required to maintain these limits. The small break LOCA analysis requires two trains of AFW to lower RCS pressure below the shutoff head of the SI pumps.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In addition to its accident mitigation function, the energy and mass addition capability of the AFW System is also considered with respect to HELBs within containment. For SLBs and FWLBs within containment, pump runout from all three AFW pumps is assumed for 10 minutes until operations can isolate the flow by tripping the AFW pumps or by closing the respective pump discharge flow path(s). Therefore, the motor operated discharge isolation valves for the motor MDAFW pump trains (4007 and 4008) are designed to limit flow to < 230 gpm. The TDAFW train is assumed to be at runout conditions (i.e., 600 gpm).

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary or containment.

The AFW System is comprised of two systems which are configured into five trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the SGs are OPERABLE (see Figures B 3.7.5-1 and 3.7.5-2). This requires that the following be OPERABLE:

- a. Two MDAFW trains taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), and capable of supplying their respective SG with ≥ 200 gpm and ≤ 230 gpm total flow;
- b. The TDAFW train taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), provided steam is available from both main steam lines upstream of the MSIVs, and capable of supplying both SGs with ≥ 200 gpm each; and

(continued)

BASES

LCO
(continued)

- c. Two motor driven SAFW trains capable of being initiated either locally or from the control room within 10 minutes, taking suction from the SW System, and supplying their respective SG and the opposite SG through the SAFW cross-tie line with ≥ 200 gpm.

The piping, valves, instrumentation, and controls in the required flow paths are also required to be OPERABLE. The TDAFW train is comprised of a common pump and two flow paths. A TDAFW train flow path is defined as the steam supply line and the SG injection line from/to the same SG. The failure of the pump or both flow paths renders the TDAFW train inoperable.

The cross-tie line for the preferred MDAFW pumps is not required for this LCO. The recirculation lines for the preferred AFW system and SAFW system pumps are not credited in the accident analysis and are also not required to be OPERABLE for this LCO since the MSSVs maintain the SG pressure below the pump's shutoff head.

The SAFW Pump Building room coolers are required to be OPERABLE when the outside air temperature is $\geq 80^\circ\text{F}$. If one room cooler is inoperable, the associated SAFW train is inoperable.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW System is lost. In addition, the AFW System is required to supply enough makeup water to replace the lost SG secondary inventory as the plant cools to MODE 4 conditions.

In MODE 4, 5, or 6, the SGs are not normally used for heat removal, and the AFW System is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If one of the TDAFW train flow paths is inoperable, action must be taken to restore the flow path to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE turbine driven AFW pump flow path;
- b. The availability of redundant OPERABLE MDAFW and SAFW pumps; and
- c. The low probability of an event occurring that requires the inoperable TDAFW pump flow path.

A TDAFW train flow path is defined as the steam supply line and SG injection line from/to the same SG.

B.1

If one MDAFW train is inoperable, action must be taken to restore the train to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE MDAFW train;
- b. The availability of redundant OPERABLE TDAFW and SAFW pumps; and
- c. The low probability of an event occurring that requires the inoperable MDAFW train.

(continued)

BASES

ACTIONS
(continued)

C.1

With the TDAFW train inoperable, or both MDAFW trains inoperable, or one TDAFW train flow path and one MDAFW train inoperable to opposite SGs, action must be taken to restore OPERABLE status within 72 hours. If the inoperable MDAFW train supplies the same SG as the inoperable TDAFW flow path, Condition D must be entered.

The combination of failures which requires entry into this Condition all result in the loss of one train (or one flow path) of preferred AFW cooling to each SG such that redundancy is lost. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

D.1

With all AFW trains to one or both SGs inoperable, action must be taken to restore at least one train or TDAFW flow path to each affected SG to OPERABLE status within 4 hours.

The combination of failures which require entry into this Condition all result in the loss of preferred AFW cooling to at least one SG. If a SGTR were to occur in this condition, preferred AFW is potentially unavailable to the unaffected SG. If AFW is unavailable to both SGs, the accident analyses for small break LOCAs and loss of MFW would not be met.

(continued)

BASES

ACTIONS

D.1 (continued)

The two MDAFW trains of the preferred AFW System are normally used for decay heat removal during low power operations since air operated bypass control valves are installed in each train to better control SG level (see Figure B 3.7.5-1). Since a feedwater transient is more likely during reduced power conditions, 4 hours is provided to restore at least one train of additional preferred AFW before requiring a controlled cooldown. This will also provide time to find a condensate source other than the SW System for the SAFW System if all three AFW trains are inoperable. The 4 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

E.1

With one SAFW train inoperable, action must be taken to restore OPERABLE status within 14 days. This Condition includes the inoperability of one of the two SAFW cross-tie valves which requires declaring the associated SAFW train inoperable (e.g., failure of 9703B would result in declaring SAFW train D inoperable). The 14 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

F.1

With both SAFW trains inoperable, action must be taken to restore at least one SAFW train to OPERABLE status within 7 days. This Condition includes the inoperability of the SAFW cross-tie. The 7 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

(continued)



BASES

ACTIONS
(continued)

G.1 and G.2

When Required Action A.1, B.1, C.1, D.1, E.1, or F.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

H.1

If all three preferred AFW trains and both SAFW trains are inoperable the plant is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one MDAFW, TDAFW, or SAFW train to OPERABLE status. For the purposes of this Required Action, only one TDAFW train flow path and the pump must be restored to exit this Condition.

Required Action H.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW and SAFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Periodically comparing the reference differential pressure and flow of each AFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.7.5.3

Periodically comparing the reference differential pressure and flow of each SAFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. Because it is undesirable to introduce SW into the SGs while they are operating, this testing is performed using the test condensate tank. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

SR 3.7.5.4

This SR verifies that each AFW and SAFW motor operated suction valve from the SW System (4013, 4027, 4028, 9629A, and 9629B), each AFW and SAFW discharge motor operated valve (4007, 4008, 9704A, 9704B, and 9746), and each SAFW cross-tie motor operated valve (9703A and 9703B) can be operated when required. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 4).

SR 3.7.5.5

This SR verifies that AFW can be delivered to the appropriate SG in the event of any accident or transient that generates an actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.5.6

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month Frequency is based on the potential need to perform this Surveillance under the conditions that apply during a plant outage.

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions may have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.7

This SR verifies that the SAFW System can be actuated and controlled from the control room. The SAFW System is assumed to be manually initiated within 10 minutes in the event that the preferred AFW System is inoperable. The Frequency of 24 months is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed at power.

REFERENCES

1. UFSAR, Section 10.5.
 2. UFSAR Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
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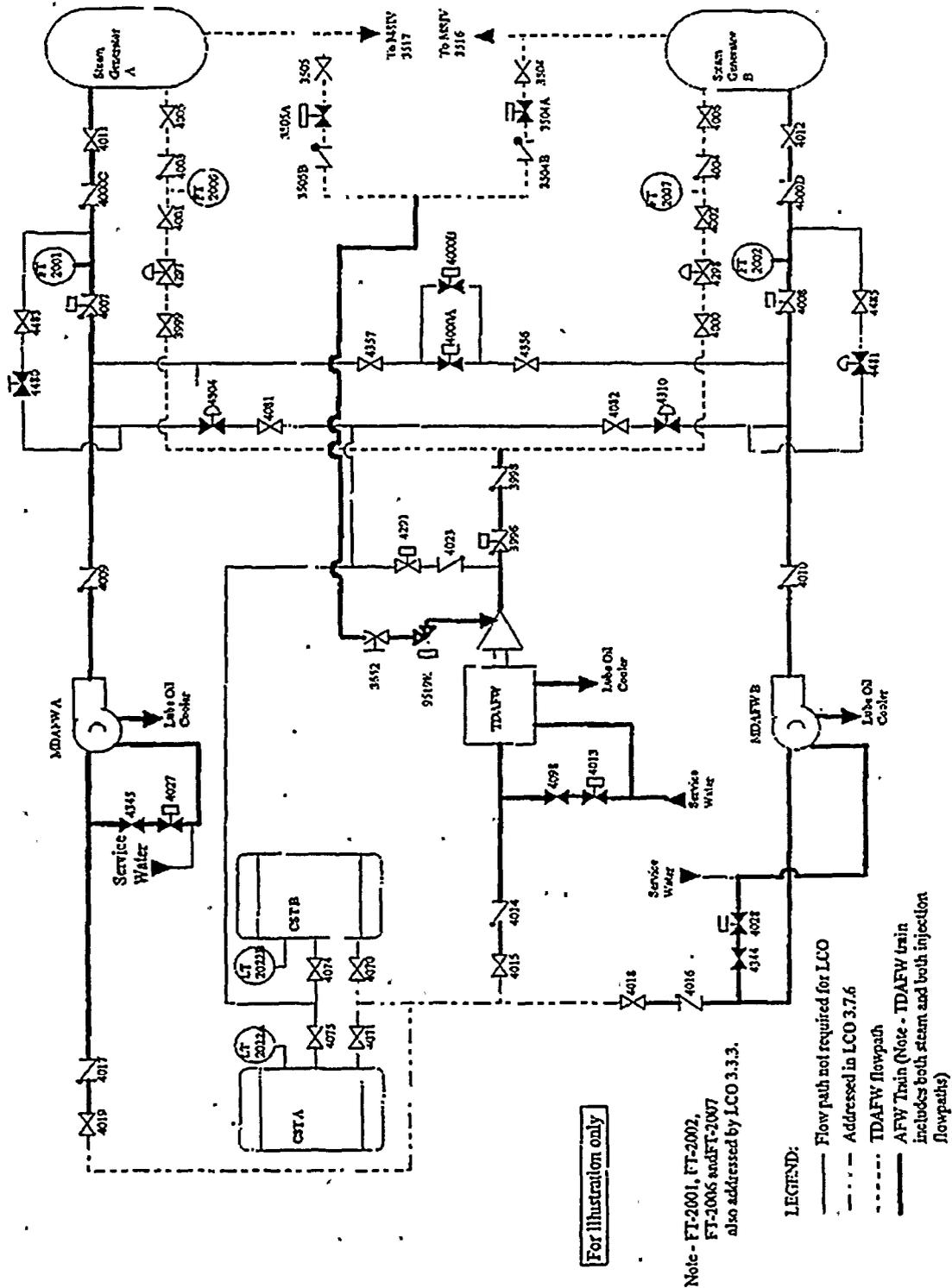
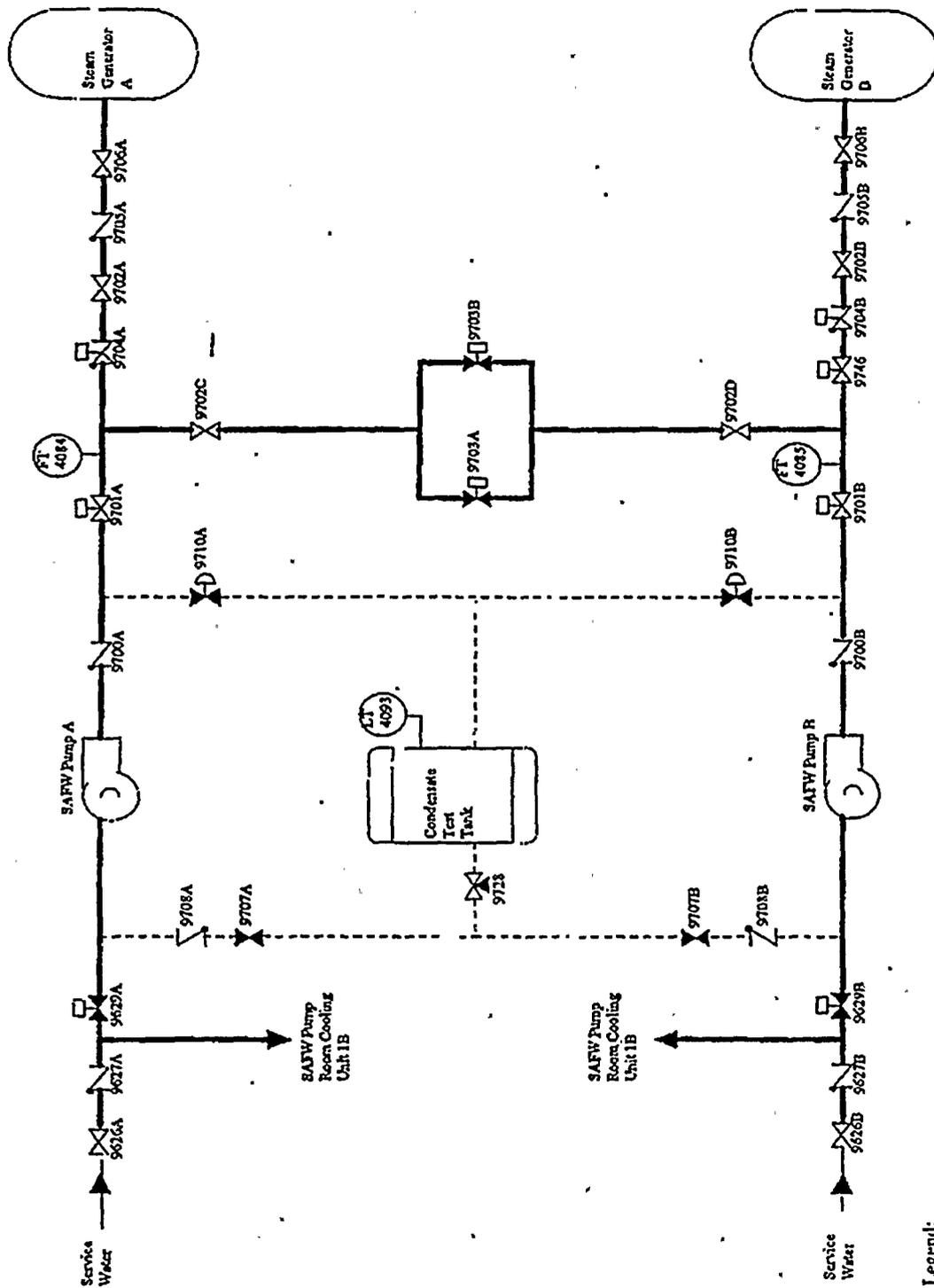


Figure B 3.7.5-1
Preferred AFW System





[For illustration only]

Legend:
 - - - - - Flow path not required for LCO
 — SAFW Train

Figure B 3.7.5-2
Standby AFW System

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tanks (CSTs)

BASES

BACKGROUND

The CSTs provide a source of water to the steam generators (SGs) for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the preferred Auxiliary Feedwater (AFW) System (LCO 3.7.5) (see Figure B 3.7.5-1). The resulting steam produced in the SGs is released to the atmosphere by the main steam safety valves or the atmospheric relief valves.

When the main steam isolation valves are open, the preferred means of heat removal from the RCS is to discharge steam to the condenser by the nonsafety grade path of the steam dump valves. The condensed steam is then returned to the SGs by the main feedwater system. This has the advantage of conserving condensate while minimizing releases to the environment.

There are two 30,000 gallon CSTs located in the non-seismic Service Building (Ref. 1). The CSTs are not considered safety related components since the tanks are not protected against earthquakes or other natural phenomena, including missiles. The safety related source of condensate for the AFW and Standby AFW Systems is the Service Water (SW) System (LCO 3.7.8). The CSTs are connected by a common header which leads to the suction of all three AFW pumps. A single level transmitter is provided for each CST (LT-2022A and LT-2022B). The CSTs can be refilled from the condenser hotwell or the all-volatile-treatment condensate storage tank.

APPLICABLE SAFETY ANALYSES

The CSTs provide cooling water to remove decay heat and to cooldown the plant following all events in the accident analysis (Ref. 2) which assumes that the preferred AFW System is available immediately following an accident. For any event in which AFW is not required for at least 10 minutes following the accident, the SW System provides the source of cooling water to remove decay heat.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting Design Basis Accident (DBA) for the condensate volume is the loss of normal feedwater event and small break loss of coolant accident (LOCA) (Ref. 2). For the loss of normal feedwater event, flow from at least two AFW pumps is required upon a low level signal in either SG to meet the acceptance criteria for a Condition 2 event (Ref. 3). For the small break LOCA, two AFW pumps are required to lower the RCS pressure below the shutoff head of the safety injection pumps. Assuming that all three AFW pumps initiate at their maximum flowrate, the CSTs provide sufficient inventory for at least 20 minutes (at greater than required flowrates) before operator action to refill the CSTs or transfer suction to the SW System is required.

A nonlimiting event considered in CST inventory determinations is a main feedwater line break inside containment. This break has the potential for dumping condensate until terminated by operator action after 10 minutes since there is no automatic re-configuration of the AFW System. Following termination of the AFW flow to the affected SG by closing the AFW train discharge valves or stopping a pump, flow from the remaining AFW train or the SAFW System is directed to the intact SG for decay heat removal. This loss of condensate is partially compensated for by the retention of inventory in the intact SG.

For cooldowns following loss of all onsite and offsite AC electrical power, the CSTs contain sufficient inventory to provide a minimum of 2 hours of decay heat removal as required by NUREG-0737 (Ref. 4), item II.E.1.1. This beyond DBA requirement provides more limiting criteria for CST inventory.

The CSTs satisfy Criterion 3 of the NRC Policy Statement.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for at least 10 minutes following a loss of MFW event from 102% RTP. After this time period, the accident analyses assume that AFW pump suction can be transferred to the safety related suction source (i.e., the SW System).

(continued)

BASES

LCO
(continued)

The required CST water volume is $\geq 22,500$ gallons, which is based on the need to provide at least 2 hours of decay heat removal following loss of all AC electrical power. The CSTs are considered OPERABLE when at least 22,500 gallons of water is available. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. Since the CSTs are 30,000 gallon tanks, only one CST is required to meet the minimum required water volume for this LCO.

The OPERABILITY of the CSTs is determined by maintaining the tank level at or above the minimum required water volume.

APPLICABILITY

In MODES 1, 2, and 3, the CSTs are required to be OPERABLE to support the AFW System requirements.

In MODE 4, 5, or 6, the CST is not required because the AFW System is not required to be OPERABLE.

ACTIONS

A.1 and A.2

If the CST water volume is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the preferred AFW pumps are OPERABLE and immediately available upon AFW initiation, and that the backup supply has the required volume of water available. Alternate sources of water include, but is not limited to, the SW System and the all-volatile-treatment condensate tank. In addition, the CSTs must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Continued verification of the backup supply is not required due to the large volume of water typically available from these alternate sources. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the backup supply cannot be verified or the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 22,500 gal minimum volume is met if one CST is ≥ 21 ft or if both CSTs are ≥ 12.5 ft. The 12 hour Frequency is based on operating experience and the need for operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. UFSAR, Section 10.7.4.
 2. UFSAR, Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
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B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the CCW System also provides this function for various safety related and nonsafety related components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water (SW) System, and thus to the environment. The safety related functions of the CCW system are covered by this LCO.

The CCW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump and heat exchanger trains (Ref. 1) (see Figure B 3.7.7-1). Each CCW train consists of a manual suction and discharge valve, a pump, and a discharge check valve. The trains discharge to a common header which then supplies two heat exchangers, either of which can supply the safety related and non-safety related components cooled by CCW. The CCW loop header begins at the common piping at the discharge of the two parallel heat exchangers, and continues up to the first isolation valve for each component supplied by the CCW System. The CCW loop header then continues from the last isolation valve on the discharge of each supplied load to the common piping at the suction of the CCW pumps. Each pump is powered from a separate Class 1E electrical bus. An open surge tank in the system provides for thermal expansion and contraction of the CCW system and ensures that sufficient net positive suction head is available to the pumps. The CCW System is also provided with a radiation detector (R-17) to isolate the surge tank from the Auxiliary Building environment and to provide indication of a leak of radioactive water into the CCW System.

The CCW System is normally maintained below 100°F by the use of one pump train in conjunction with one heat exchanger. The standby CCW pump will automatically start if the system pressure falls to 50 psig.

(continued)

BASES

BACKGROUND
(continued)

The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Since the removal of decay heat via the RHR System is only performed during the recirculation phase of an accident, the CCW pumps do not receive an automatic start signal. Following the generation of a safety injection signal, the normally operating CCW pump will remain in service unless an undervoltage signal is present on either Class 1E electrical Bus 14 or Bus 16 at which time the pump is stripped from its respective bus. A CCW pump can then be manually placed into service prior to switching to recirculation operations which would not be required until a minimum of 46 minutes following an accident.

APPLICABLE
SAFETY ANALYSES

The design basis of the CCW System is for one CCW train and one CCW heat exchanger to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase. The Emergency Core Cooling System (ECCS) and containment models for a LOCA each consider the minimum performance of the CCW System. The normal temperature of the CCW is ≤ 100 °F, and, during LOCA conditions, a maximum temperature of 120°F is assumed. This prevents the CCW System from exceeding its design temperature limit of 200°F, and provides for a gradual reduction in the temperature of containment sump fluid as it is recirculated to the Reactor Coolant System (RCS) by the ECCS pumps. The CCW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

The CCW trains, heat exchangers, and loop headers are manually placed into service prior to the recirculation phase of an accident (i.e., 46 minutes following a large break LOCA).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The CCW System can also function to cool the plant from RHR entry conditions ($T_{avg} < 350^{\circ}\text{F}$), to MODE 5 ($T_{avg} < 200^{\circ}\text{F}$), during normal cooldown operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR trains operating. Since CCW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in draining the CCW System within a short period of time. The CCW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of CCW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the Auxiliary Feedwater System) with acceptable results (Ref. 1). Leaks within the CCW System during post accident conditions can be mitigated by the available makeup water sources.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one CCW train, one heat exchanger, and the loop header is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water (see Figure B 3.7.7-1). To ensure this requirement is met, two trains of CCW, two heat exchangers, and the loop header must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCW train is considered OPERABLE when the pump is OPERABLE and capable of providing cooling water to the loop header. The automatic start logic associated with low CCW system pressure is not required for this LCO. In addition, if a CCW pump fails an Inservice Testing Program surveillance (e.g., pump developed head) the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses.

(continued)

BASES

LCO
(continued)

The CCW loop header is considered OPERABLE when the associated piping, valves, surge tank, and the instrumentation and controls required to provide cooling water to the following safety related components are available and capable of performing their safety related function:

- a. Two RHR heat exchangers;
- b. Two RHR pump mechanical seal coolers and bearing water jackets;
- c. Three safety injection pump mechanical seal coolers; and
- d. Two containment spray pump mechanical seal coolers.

The CCW loop header temperature must also be $\leq 120^{\circ}\text{F}$ prior to the CCW cooling water reaching the first isolation valve supplying these components.

The CCW loop header begins at the common piping at the discharge of the CCW heat exchangers and continues up to the first isolation valve for each of the above components. The CCW loop header then continues from the last isolation valve on the discharge of each of the above components to the common piping at the suction of the CCW pumps.

The portion of CCW piping, valves, instrumentation and controls between the isolation valves to components a through d above is addressed by the following LCOs:

- a. LCO 3.4.6, "RCS Loops - MODE 4,"
- b. LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
- c. LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"

(continued)

BASES

LCO
(continued)

- d. LCO 3.5.2, "ECCS - MODES 1, 2, and 3,"
- e. LCO 3.5.3, "ECCS - MODE 4,"
- f. LCO 3.9.3, "RHR and Coolant Circulation - Water Level \geq 23 Ft," and
- g. LCO 3.9.4, "RHR and Coolant Circulation - Water Level $<$ 23 Ft."

The CCW piping inside containment for the reactor coolant pumps (RCPs) and the reactor support coolers also serves as a containment isolation boundary. This is addressed by LCO 3.6.3, "Containment Isolation Boundaries."

The CCW system radiation detector (R-17) is not required to be OPERABLE for this LCO since the CCW system outside containment is not required to be a closed system.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be capable to perform its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling and containment integrity during the recirculation phase following a LOCA.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by LCO 3.4.7, LCO 3.4.8, LCO 3.9.3, and LCO 3.9.4.

(continued)



BASES (continued)

ACTIONS

A.1

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE CCW train could result in loss of CCW function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

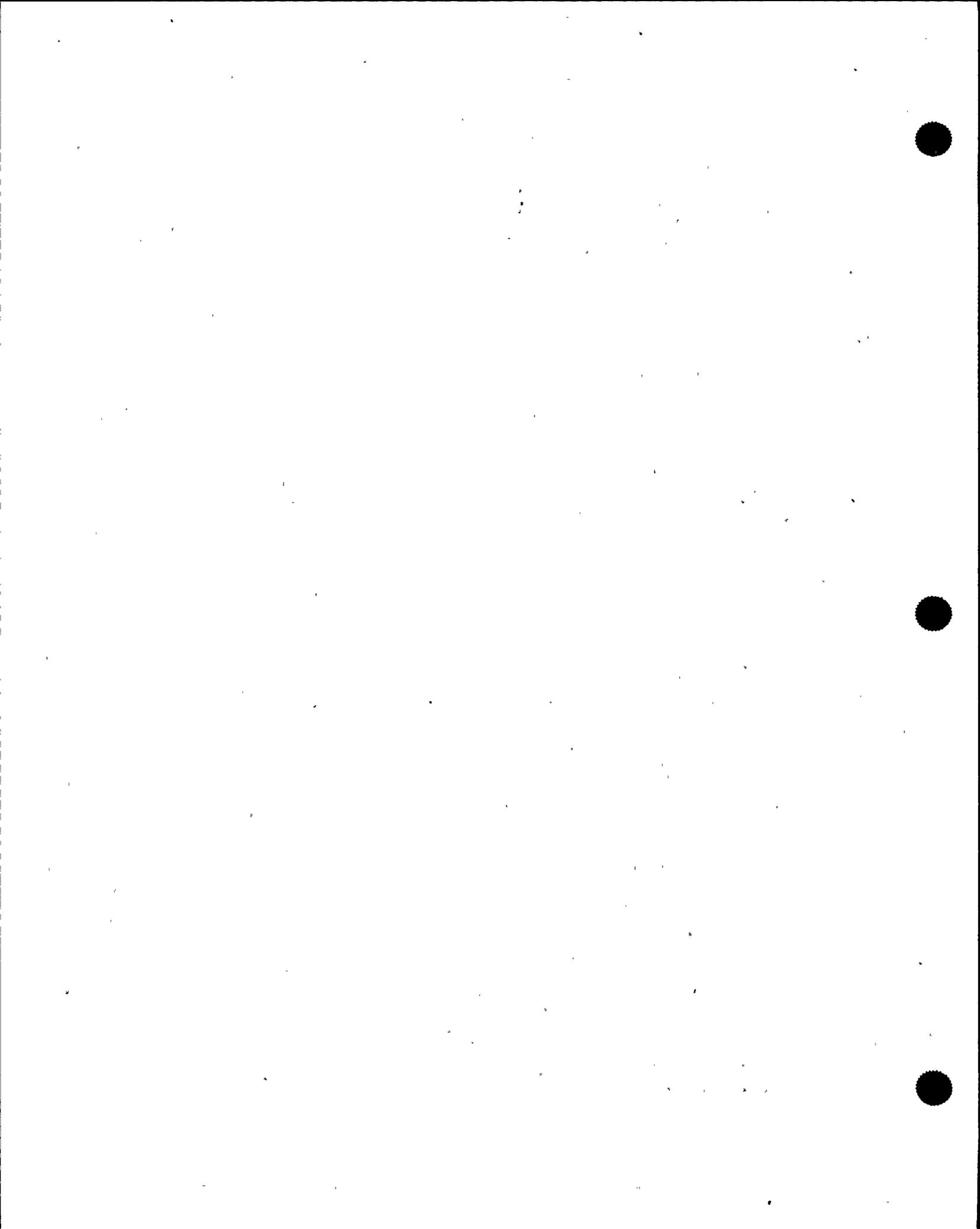
B.1

If one CCW heat exchanger is inoperable, action must be taken to restore OPERABLE status within 31 days. In this Condition, the remaining OPERABLE heat exchanger is adequate to perform the heat removal function. However, the overall reliability is reduced because a passive failure in the OPERABLE CCW heat exchanger could result in a loss of CCW function. The 31 day Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a passive failure of the remaining heat exchanger.

C.1 and C.2

If the CCW train or CCW heat exchanger cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

ACTIONS
(continued)

D.1, D.2, and D.3

With both CCW trains, both CCW heat exchangers, or the loop header inoperable, action must be immediately initiated to restore OPERABLE status to one CCW train, one CCW heat exchanger, and the loop header. In this Condition, there is no OPERABLE CCW System available to provide necessary cooling water which is a loss of a safety function. Also, the plant must be placed in a MODE in which the consequences of a loss of CCW coincident with an accident are reduced. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The plant is not required to exit the Applicability for this LCO (i.e., enter MODE 5) until at least one CCW train, one CCW heat exchanger, and the loop header is restored to OPERABLE status to support RHR operation.

Required Actions D.1, D.2, and D.3 are modified by a Note indicating that all required MODE changes or power reductions required by other LCOs are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual and power operated valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW loop header.

SR 3.7.7.2

This SR verifies that the two motor operated isolation valves to the RHR heat exchangers (738A and 738B) can be operated when required since the valves are normally maintained closed. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 2).

REFERENCES

1. UFSAR, Section 9.2.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
-

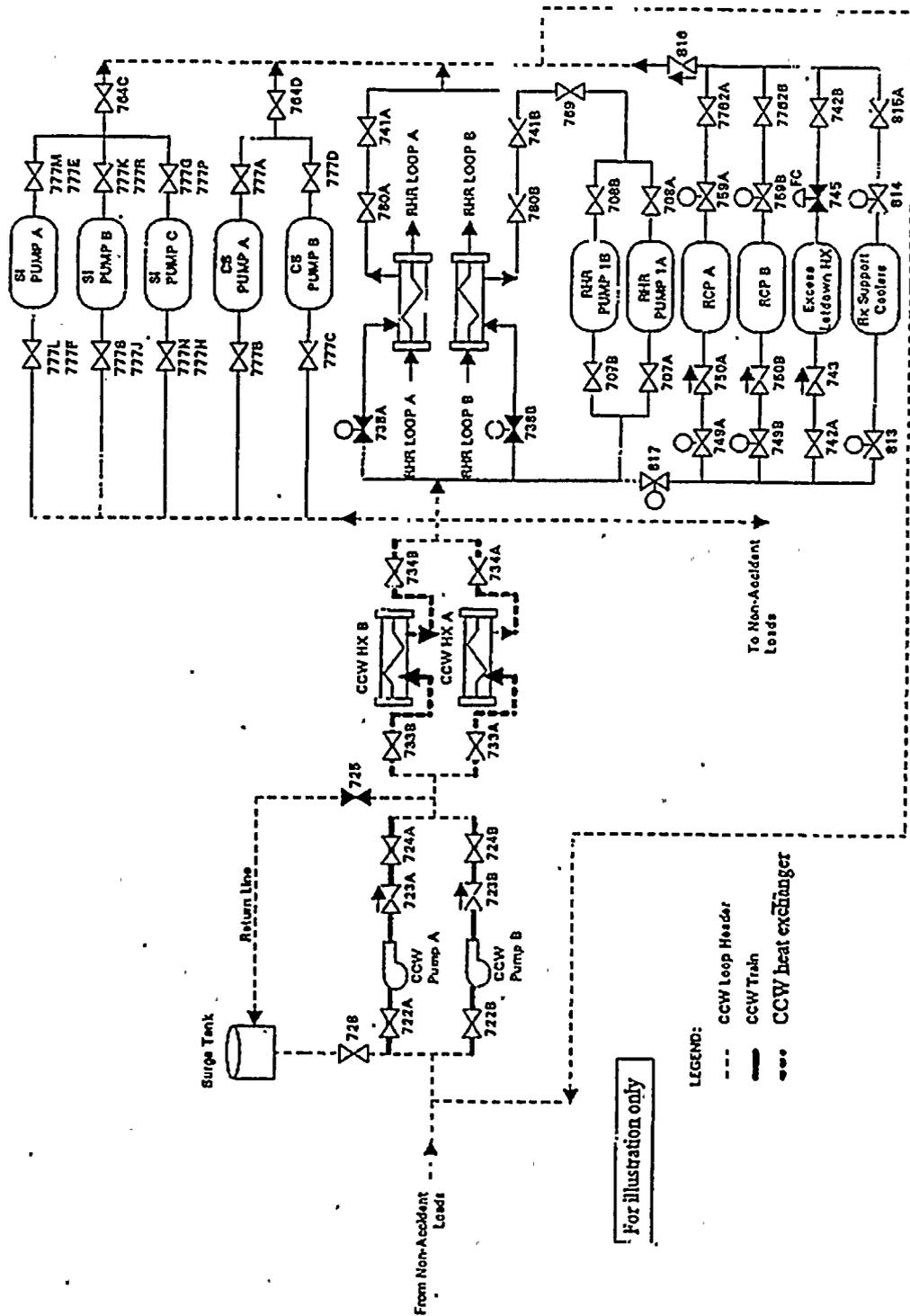


Figure B 3.7.7-1
CCW System

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW system also provides this function for various safety related and nonsafety related components. The safety related functions of the SW System are covered by this LCO.

The SW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump trains (Ref. 1) (see Figure B 3.7.8-1). The physical design of the SW System is such that one 100% capacity pump from each class 1E electrical bus (Buses 17 and 18) is arranged on a common piping header which then supplies the SW loop header. For the purposes of this LCO, a SW train is based on electrical source only.

Each train is powered from a separate Class 1E electrical bus and consists of two 100% capacity pumps and associated discharge check valves and manual isolation valves. The SW loop header begins from the discharge of the trains and supplies the safety related and nonsafety related components cooled by SW. The pumps in the system are normally manually aligned. One pump in each train is selected to automatically start upon receipt of an undervoltage signal on its respective bus. Upon receipt of a safety injection signal, each SW pump will automatically start in a predetermined sequence.

The SW loop header supplies the cooling water to all safety related and nonsafety related components. The nonsafety related and long-term safety functions (e.g., component cooling water heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. These valves automatically close on a coincident safety injection signal and undervoltage signal on Buses 14 and 16.

(continued)

BASES

BACKGROUND
(continued)

The suction source for the SW System is the screenhouse which is a seismic structure located on Lake Ontario. The discharge from the SW System supplied loads returns back to Lake Ontario. The principal safety related functions of the SW system is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System, provide cooling water to the diesel generators (DGs) and containment recirculation fan coolers (CRFCs) and to provide a safety related source of water to the Auxiliary Feedwater (AFW) System.

APPLICABLE
SAFETY ANALYSES

The design basis of the SW System is for one SW train in conjunction with a 100% capacity containment cooling system (i.e., CRFC) to provide for heat removal following a steam line break (SLB) inside containment to ensure containment integrity. The SW System is also designed, in conjunction with the CCW System and a 100% capacity Emergency Core Cooling System and containment cooling system, to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is recirculated to the Reactor Coolant System by the ECCS pumps. The SW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

Following the receipt of a safety injection signal, all four SW pumps are designed to start (if not already running) to supply the system loads. If a coincident safety injection and undervoltage signal occurs, then each nonsafety related and nonessential load within the SW System is isolated by redundant motor operated valves that are powered by separate Class 1E electrical trains. The SW pumps are sequenced to start within 17 seconds following a safety injection signal. The selected SW pumps are sequenced to start after a 40 second time delay following energization of the electrical bus supplying the selected pump (i.e., Bus 17 or Bus 18) after an undervoltage signal.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SW trains and loop header are assumed to supply to following components following an accident:

- a. The CRFCs, DGs and safety injection pump bearing housing coolers immediately following a safety injection signal (i.e., after the loop header becomes refilled);
- b. The preferred AFW and SAFW pumps within 10 minutes following receipt of a low SG level signal; and
- c. The CCW heat exchangers within 46 minutes following a safety injection signal.

The SW system, in conjunction with the CCW System, can also cool the plant from residual heat removal (RHR) entry conditions ($T_{avg} < 350^{\circ}\text{F}$) to MODE 5 ($T_{avg} < 200^{\circ}\text{F}$) during normal operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR System trains that are operating. Since SW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in failing the SW System to potentially multiple safety related functions. The SW system has been evaluated to demonstrate the capability to meet cooling needs with an assumed 500 gal leak. The SW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of SW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the AFW Systems) with acceptable results (Ref. 1).

The temperature of the fluid supplied by the SW System is also a consideration in the accident analyses. If the cooling water supply to the containment recirculation fan coolers and CCW heat exchangers is too warm, the accident analyses with respect to containment pressure response following a SLB and the containment sump fluid temperature following a LOCA may no longer be bounding. If the cooling water supply is too cold, the containment heat removal systems may be more efficient than assumed in the accident analysis. This causes the backpressure in containment to be reduced which potentially results in increased peak clad temperatures.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one SW train and the loop header is required to be OPERABLE to provide the minimum heat removal capability to ensure that the system functions to remove post accident heat loads as assumed in the safety analyses. To ensure this requirement is met, two trains of SW and the loop header must be OPERABLE (see Figure B 3.7.8-1). At least one SW train will operate assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW train is defined based on electrical power source such that SW Pumps A and C form one train and SW Pumps B and D form the second train. A SW train is considered OPERABLE when one pump in the train is OPERABLE and capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW pumps:

- a. Level \geq 5 feet; and
- b. Temperature \geq 35°F above 50% RTP and \leq 80°F.

The lower screenhouse bay temperature is only specified above 50% RTP since this value is only a consideration when evaluating LOCA at or near full power conditions. In addition, if a SW pump fails on Inservice Testing Program surveillance (e.g., pump developed head), the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses (Ref. 1).

(continued)

BASES

LCO
(continued)

An OPERABLE SW train also requires that all nonessential and nonsafety related loads can be isolated by the six motor operated isolation valves which are powered from the same Class 1E electrical train as the pumps. Therefore, motor operated valves 4609, 4614, 4615, 4616, 4663, and 4670 must be OPERABLE and capable of closing for SW Pumps A and C while valves 4613, 4664, 4733, 4734, 4735, and 4780 must be OPERABLE and capable of closing for SW Pumps B and D.

The SW loop header is considered OPERABLE when the associated piping, valves, and the instrumentation and controls required to provide cooling water from each OPERABLE SW train to the following safety related components are available and capable of performing their safety related function:

- a. Four CRFCs;
- b. Two CCW heat exchangers;
- c. Two DGs;
- d. Three preferred AFW pumps;
- e. Two standby AFW pumps; and
- f. Three safety injection pump bearing housing coolers.

An OPERABLE SW loop header also requires a flow path through the diesel generator (4665, 4760, 4669, and 4668B) and CRFC (4623, 4640, 4756 and 4639) cross-ties.

The SW loop header begins at the common piping at the discharge of both SW pump trains and ends at the first isolation valve for each of the above components. Since the SW System discharges back to Lake Ontario, the cooling water flow path through the above components and subsequent discharge is addressed under their respective LCO. This includes:

- a. LCO 3.5.2, "ECCS - MODES 1, 2, and 3;"
- b. LCO 3.5.3, "ECCS - MODE 4;"

(continued)

BASES

LCO
(continued)

- c. LCO 3.6.6, "CS, CRFC, and Post-Accident Charcoal Systems;"
- d. LCO 3.7.5, "AFW Systems;"
- e. LCO 3.7.7, "CCW System;"
- f. LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4;" and
- g. LCO 3.8.2, "AC Sources - MODES 5 and 6."

The SW piping inside containment for the CRFCs and the reactor compartment coolers also serves as a containment isolation boundaries. This is addressed under LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABILITY

In MODES 1, 2, 3, and 4, the SW System is a normally operating system which must be capable of performing its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling during the recirculation phase following a LOCA or loss of containment integrity following a SLB.

In MODES 5 and 6, the OPERABILITY requirements of the SW system are determined by LCO 3.6.6, LCO 3.7.7, and LCO 3.8.2.

ACTIONS

A.1

If one SW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW train could result in loss of SW System function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the SW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

With both SW trains or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

Required Action C.1 is modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be entered for the component cooling water heat exchanger made inoperable by SW. This note is provided since the inoperable SW system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.

SURVEILLANCE
REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.8.2

Verifying the correct alignment for manual, power operated, and automatic valves in the SW flow path provides assurance that the proper flow paths exist for SW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the SW flow to individual components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

SR 3.7.8.3

This SR verifies that all SW loop header cross-tie valves are locked in the correct position. This includes verification that manual valves 4623, 4639, 4640, 4665, 4668B, 4669, 4756, and 4760 are locked open and that manual valves 4610, 4611, 4612, and 4779 are locked closed. The 31 day Frequency is based on engineering judgement, is consistent with the procedural controls governing locked valves, and ensures correct valve positions.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.8.4

This SR verifies proper automatic operation of the SW motor operated isolation valves on an actual or simulated actuation signal (i.e., coincident safety injection and undervoltage signal). SW is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.5

This SR verifies proper automatic operation of the SW pumps on an actual or simulated actuation signal. This includes the actuation of the SW pumps following an undervoltage signal and following a coincident safety injection and undervoltage signal. SW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.1.
 2. UFSAR, Section 6.2.
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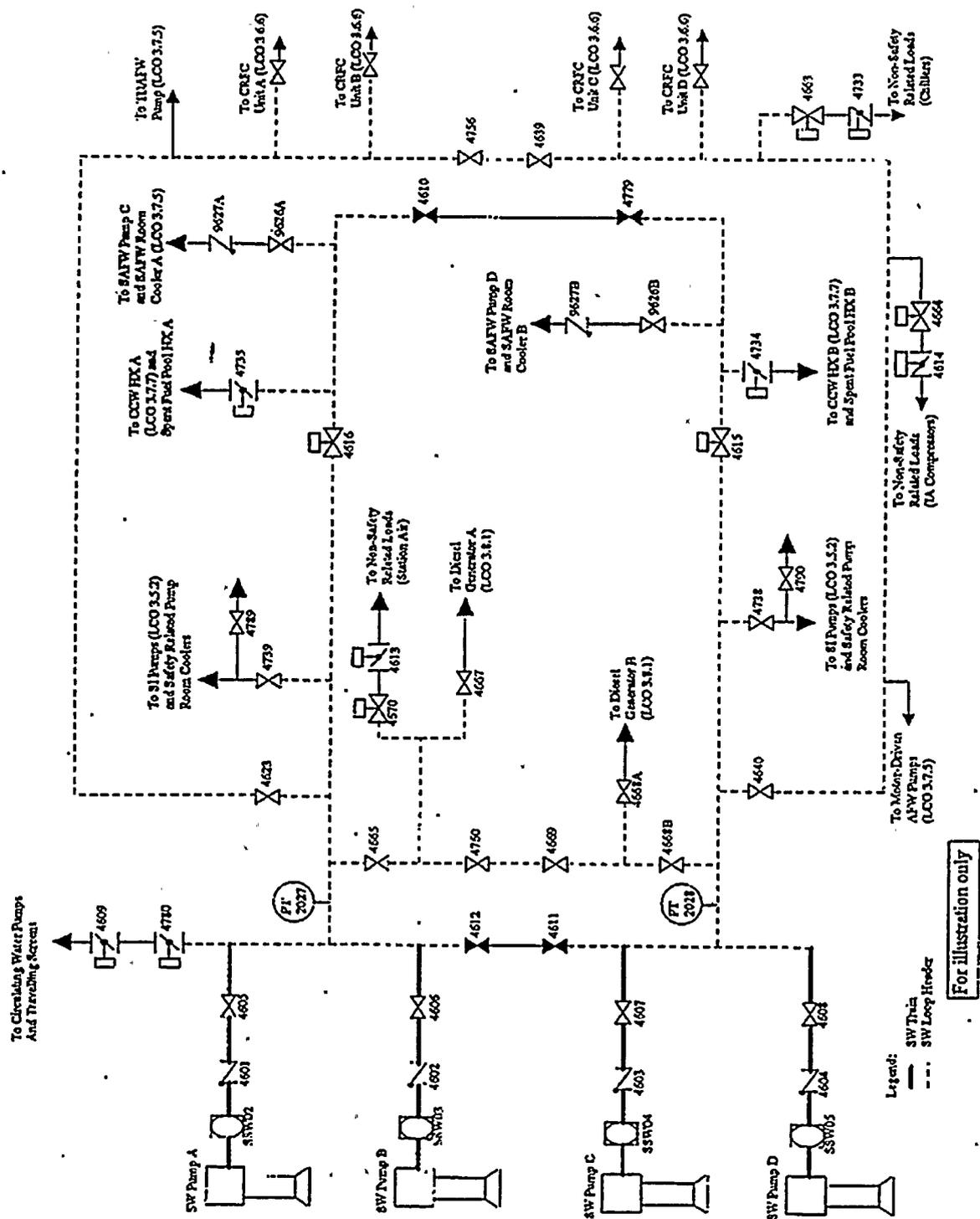


Figure B 3.7.8-1
SW System

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Air Treatment System (CREATS)

BASES

BACKGROUND

According to Atomic Industry Forum (AIF) GDC 11 (Ref. 1), a control room shall be provided which permits continuous occupancy under any credible postaccident condition without excessive radiation exposures of personnel. Exposure limits are provided in GDC 19 of 10 CFR 50, Appendix A (Ref. 2) which requires that control room personnel be restricted to 5 rem whole body, or its equivalency, for the duration of the accident. The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity for 30 days without exceeding this 5 rem whole body limit. The CREATS is part of the Control Building ventilation system.

The CREATS consists of a high efficiency particulate air (HEPA) filter, activated charcoal adsorbers for removal of gaseous activity (principally iodines), and two fans (control room return air fan and emergency return air fan) (see Figure B 3.7.9-1). Ductwork, dampers, and instrumentation also form part of the system as well as demisters to remove water droplets from the air stream (Ref. 3).

The CREATS is an emergency system, parts of which may operate during normal plant operations. Actuation of the CREATS places the system in one of five separate states of the emergency mode of operation, depending on the initiation signal. The following are the normal and emergency modes of operation for the CREATS:

CREATS Mode A

The CREATS is in the standby mode with the exception that the control room return air fan is in operation.

(continued)

BASES

BACKGROUND
(continued)

CREATS Mode B

This is the CREATS configuration following an accident with a radiation release as detected by radiation monitor R-1. Upon receipt of an actuation signal, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm (approximately one fourth of the Control Building Ventilation System design) through the CREATS charcoal and HEPA filters. All outside air that enters the CREATS, as controlled by an air adjust switch (S-81), is also circulated through the CREATS charcoal and HEPA filters.

CREATS Mode C

This is the same CREATS configuration as Mode B with the exception that all outside air is isolated to the control room by one damper in each air supply flow path.

CREATS Mode D

This is the CREATS configuration following the detection of smoke within the Control Building. Upon receipt of an actuation signal, the system continues to draw outside air. However, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm through the CREATS and HEPA filters. This effectively purges the control room air environment.

CREATS Mode E

This is the same CREATS configuration as Mode D with exception that all outside air is isolated to the control room by one damper in each air supply flow path.

(continued)

BASES

BACKGROUND
(continued)CREATS Mode F

This is the CREATS configuration following the detection of a toxic gas as indicated by the chlorine or ammonia detectors, or high radiation as detected by R-36 (gas), R-37 (particulate), or R-38 (iodine). Upon receipt of an actuation signal, the system aligns itself consistent with Mode C except that two dampers in each air supply path are isolated.

Normally open air supply isolation dampers are arranged in series so that the failure of one damper to close will not result in a breach of isolation.

The air entering the control room is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. The actions of the toxic gas and high radiation state (Mode F) are more restrictive, and will override the actions of the emergency radiation state (Mode B or C). Only the high radiation state CREATS Mode F is addressed by this LCO.

APPLICABLE
SAFETY ANALYSES

The location of components and CREATS related ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 3). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

In MODES 5 and 6, and during movement of irradiated fuel assemblies, the CREATS ensures control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The CREATS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The CREATS is comprised of a filtration train and two independent and redundant isolation damper trains all of which are required to be OPERABLE. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

The CREATS is considered OPERABLE when the individual components necessary to permit CREATS Mode F operation are OPERABLE (see Figure B 3.7.9-1). The CREATS filtration train is OPERABLE when the associated:

- a. Control room return air and emergency return air fans are OPERABLE and capable of providing forced flow;
- b. HEPA filters and charcoal adsorbers for the emergency return air fan are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. Ductwork, valves, and dampers (including AKD06 and AKD09) are OPERABLE, and air circulation can be maintained.

The CREATS isolation dampers are considered OPERABLE when the damper (AKD01, AKD04, AKD05, AKD08, and AKD10) can close on an actuation signal to isolate outside air or is closed with motive force removed. Two dampers are provided for each outside air path.

(continued)

BASES

LCO
(continued)

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors. Opening of the access doors for entry and exit does not violate the control room boundary. An access door may be opened for extended periods provided a dedicated individual is stationed at the access door to ensure closure, if required (i.e., the individual performs the isolation function), the door is able to be closed within 30 seconds upon indication of the need to close the door, and the CREATS filtration train is OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS must be OPERABLE to control operator exposure during and following a DBA.

In MODE 5 or 6, the CREATS is required to cope with the release from the rupture of a waste gas decay tank.

During movement of irradiated fuel assemblies, the CREATS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1 and A.2

With the CREATS filtration train inoperable, action must be taken to restore OPERABLE status within 48 hours or isolate the control room from outside air. In this Condition, the isolation dampers are adequate to perform the control room protection function but no means exist to filter the release of radioactive gas within the control room. The 48 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to isolate the control room.

Required Action A.2 is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

(continued)

BASES

ACTIONS
(continued)B.1

With one CREATS isolation damper inoperable for one or more outside air flow paths, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREATS isolation damper is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREATS isolation damper could result in loss of CREATS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining isolation damper to provide the required isolation capability.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 , D.2.1, and D.2.2

In MODE 5 or 6 or during movement of irradiated fuel assemblies, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, action must be taken to immediately place the OPERABLE isolation damper(s) in CREATS Mode F. This action ensures that the remaining damper(s) are OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

(continued)

BASES

ACTIONS

D.1 , D.2.1, and D.2.2 (continued)

An alternative to Required Action D.1 is immediately suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel or other components to a safe position.

E.1

In MODE 1, 2, 3, or 4, if both CREATS isolation dampers for one or more outside air flow paths are inoperable, the CREATS may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Failure of the integrity of the control room boundary (i.e., walls, floors, ceilings, ductwork or access doors) also results in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1, and F.2, and F.3

In MODE 5 or 6 or during movement of irradiated fuel assemblies with two CREATS isolation dampers for one or more outside air flow paths inoperable, action must be taken immediately to restore one isolation damper in each affected air supply path to OPERABLE status. In addition, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes accident risk. This does not preclude the movement of fuel or other components to a safe position.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each CREATS filtration train once every 31 days for ≥ 15 minutes provides an adequate check of this system. The 31 day Frequency is based on the reliability of the equipment.

SR 3.7.9.2

This SR verifies that the required CREATS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREATS filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The minimum required flowrate through the CREATS filtration train is 2000 cubic feet per minute ($\pm 10\%$). Specific test Frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 4).

SR 3.7.9.3

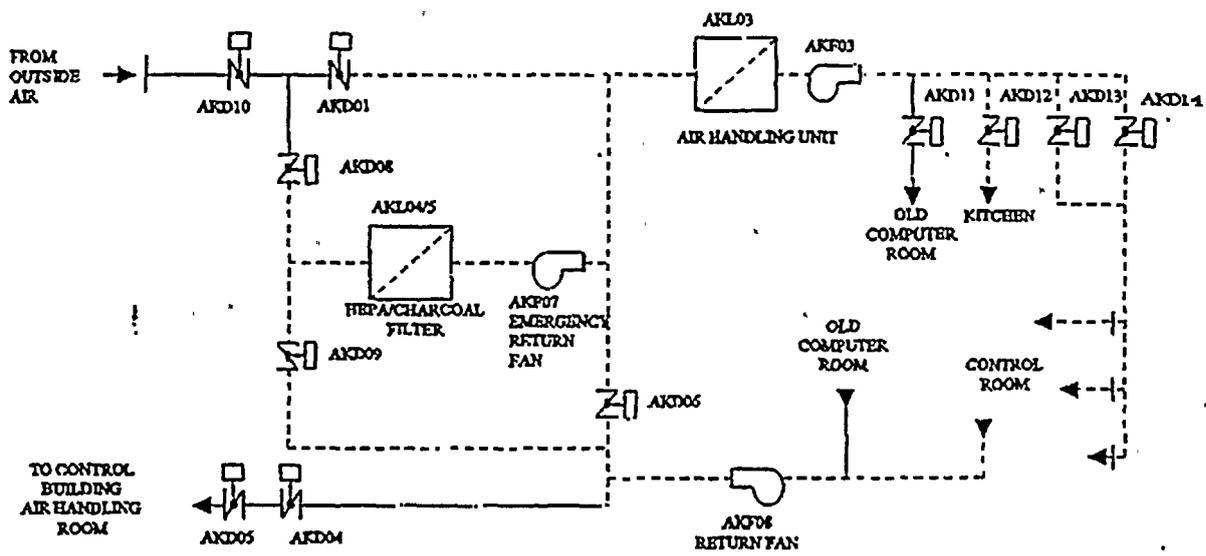
This SR verifies that the CREATS filtration train starts and operates and each CREATS isolation damper actuates on an actual or simulated actuation signal. The Frequency of 24 months is based on Regulatory Guide 1.52 (Ref. 4).

(continued)

BASES (continued)

REFERENCES

1. Atomic Industry Forum (AIF) GDC 11, Issued for comment July 10, 1967.
 2. 10 CFR 50, Appendix A, GDC 19.
 3. UFSAR, Section 6.4.
 4. Regulatory Guide 1.52, Revision 2.
-



Legend:

----- CREATS Filtration Train

For illustration only

Notes:

1. Outside air flowpath isolation dampers includes AKD01, AKD04, AKD05, AKD08, and AKD10.
2. The CREATS filtration train does not include the air handling unit (AKL03 and AKF03).

Figure B 3.7.9-1
CREATS

B 3.7 PLANT SYSTEMS

B 3.7.10 Auxiliary Building Ventilation System (ABVS)

BASES

BACKGROUND

The ABVS filters airborne radioactive particulates from the area of the spent fuel pool (SFP) following a fuel handling accident. The ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the Auxiliary Building including the SFP area.

The ABVS consists of an air handling unit, a series of exhaust fans, charcoal filters, ductwork, and dampers (Ref. 1). The exhaust fans include the following fans which all discharge into a common ductwork that supplies the Auxiliary Building main exhaust fans A and B (see Figure B 3.7.10-1):

- a. Intermediate Building exhaust fans A and B;
- b. Auxiliary Building exhaust fan C;
- c. Auxiliary Building charcoal filter fans A and B;
- d. Auxiliary Building exhaust fan G; and
- e. Control access exhaust fans A and B.

The only components which filter the environment associated with the SFP are the Auxiliary Building main exhaust fans and Auxiliary Building exhaust fan C. Therefore, these are the only fans considered with respect to the ABVS in this LCO.

(continued)

BASES

BACKGROUND
(continued)

Auxiliary Building exhaust fan C takes suction from the SFP and decontamination pit areas on the operating level of the Auxiliary Building. The air is first drawn through the SFP Charcoal Adsorber System which consists of roughing filters and charcoal adsorbers. The roughing filters protect the charcoal adsorbers from being fouled with dirt particles while the charcoal adsorbers remove the radioactive iodines from the atmosphere. Auxiliary Building exhaust fan C then discharges into the common ductwork that supplies the Auxiliary Building main exhaust fans. This common ductwork contains a high efficiency particulate air (HEPA) filter which is not credited in the dose analyses.

The Auxiliary Building main exhaust fans are each 100% capacity fans which can maintain a negative pressure on the operating floor of the Auxiliary Building through orientation of the system dampers. This negative pressure causes air flow on the operating floor to be toward the SFP which ensures that air in the vicinity of the SFP is first filtered through the SFP Charcoal Adsorber System. The Auxiliary Building main exhaust fans and exhaust fan C are powered from non-Engineered Safeguards Features buses.

The Auxiliary Building main exhaust fans discharge to the plant vent stack. The plant vent stack is continuously monitored for noble gases (R-14), particulates (R-13) and iodine (R-10B). During normal power operation, the ABVS is placed in the "out" mode by the interlock mode switch where "out" defines the status of the SFP charcoal filters. This causes all exhaust fans without any HEPA or charcoal filters (excluding the Auxiliary Building Main exhaust fans) and Auxiliary Building exhaust fan C to trip upon a signal from R-10B, R-13 or R-14 to stop the release of any radioactive gases. During fuel movement within the Auxiliary Building, the interlock mode switch is placed in the "in" mode such that only exhaust fans without any HEPA or charcoal filters (excluding Auxiliary Building main exhaust fans) are tripped.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The ABVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that Auxiliary Building exhaust fan C, the SFP Charcoal Adsorber System, and one Auxiliary Building main exhaust fan are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the minimum filtration system components which result in offsite doses well within the limits of 10 CFR 100 (Ref. 3). The failure of any or all of these filtration system components results in doses which are slightly higher but still within 10 CFR 100 limits. The fuel handling accident assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The remainder of the ABVS described in the Background is not required for any DBA since it is non-safety related and supplied only from offsite power sources.

The ABVS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The ABVS is required to be OPERABLE to ensure that offsite doses are well within the limits of 10 CFR 100 (Ref. 3) following a fuel handling accident in the Auxiliary Building. The failure of the ABVS coincident with a fuel handling accident results in doses which are slightly higher but still within 10 CFR 100 limits.

The ABVS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary Building following a fuel handling accident are OPERABLE and in operation (see Figure B 3.7.10-1). The ABVS is considered OPERABLE when its associated:

- a. Auxiliary Building exhaust fan C and either Auxiliary Building main exhaust fan A or B is OPERABLE and in operation;

(continued)

BASES

LCO
(continued)

- b. Auxiliary Building main exhaust fan HEPA filter and SFP charcoal adsorbers are not excessively restricting flow, and the SFP Charcoal Adsorber System is capable of performing its filtration function;
 - c. Ductwork, valves, and dampers are OPERABLE, and air circulation and negative pressure can be maintained on the Auxiliary Building operating floor; and
 - d. Interlock mode switch is placed in the "in" mode.
-

APPLICABILITY

During movement of irradiated fuel in the Auxiliary Building, the ABVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. The ABVS is only required when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. Any fuel handling accident which occurs after 60 days results in offsite doses which are well within 10 CFR 100 limits (Ref. 3) due to the decay rate of iodine.

Since a fuel handling accident can only occur as a result of fuel movement, the ABVS is not MODE dependant and only required when irradiated fuel is being moved.

ACTIONS

A.1

When the ABVS is inoperable, action must be taken to place the plant in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies in the Auxiliary Building which have decayed < 60 days since being irradiated, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.10.1

This SR verifies the OPERABILITY of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered LEAKAGE. This SR ensures that Auxiliary Building exhaust fan C, and either Auxiliary Building main exhaust fan A or B are in operation and that the ABVS interlock mode switch is in the correct position. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

SR 3.7.10.2

This SR verifies the integrity of the Auxiliary Building enclosure. The ability of the Auxiliary Building to maintain negative pressure with respect to the uncontaminated outside environment must be periodically verified to ensure proper functioning of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered leakage. This SR ensures that a negative pressure is being maintained in the Auxiliary Building. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

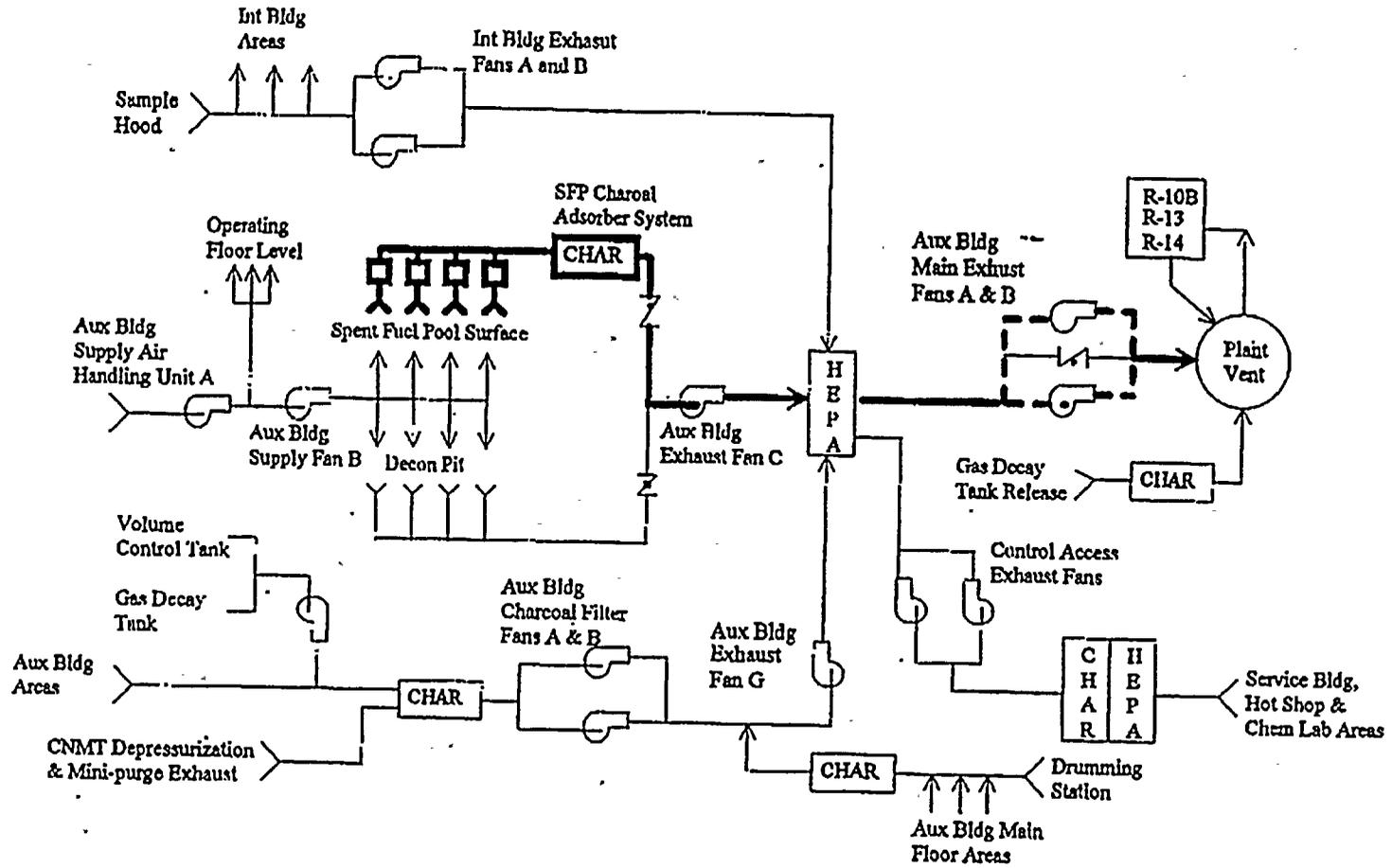
SR 3.7.10.3

This SR verifies that the required SFP Charcoal Adsorber System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SFP Charcoal Adsorber System filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). There is no minimum required flowrate through the SFP charcoal adsorbers since SR 3.7.10.2 requires verification that a negative pressure is maintained during fuel movement in the Auxiliary Building. As long as this minimum pressure is maintained by drawing air from the surface of the SFP through the SFP charcoal adsorbers, the assumptions of the accident analyses are met. Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 5).

REFERENCES

1. UFSAR, Section 9.4.2.
 2. UFSAR, Section 15.7.3.2.
 3. 10 CFR 100.
 4. Regulatory Guide 1.25, Rev. 0.
 5. Regulatory Guide 1.52, Rev. 2.
-

Figure B 3.7.10-1
ABVS



Legend:

- Flowpath required by LCO (Aux Bldg Exhaust Fan C HEPA filter not required for LCO but Aux Bldg operating floor must be at a negative pressure)
- - - 1 of 2 flowpaths required by LCO
- ▣ SFP Roughing filters

For illustration only

B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool (SFP) meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level provides protection against exceeding the offsite dose limits.

The SFP is a seismically designed structure located in the Auxiliary Building (Ref. 1). The pool is internally clad with stainless steel that has a leak chase system at each weld seam, to minimize accidental drainage through the liner. The SFP is also provided with a barrier between the spent fuel storage racks and the fuel transfer system winch. This barrier, up to the height of the spent fuel racks, prevents inadvertent drainage of the SFP via the fuel transfer tube.

The SFP Cooling System is designed to maintain the pool $\leq 120^{\circ}\text{F}$ during normal conditions and refueling operations (Ref. 2). The cooling system normally takes suction near the surface of the SFP such that a failure of any pipe in the system will not drain the pool. The cooling system return line to the pool also contains a 0.25 inch vent hole located near the SFP surface level to prevent siphoning. Finally, control board alarms exist with respect to the SFP level and temperature. These features all help to prevent inadvertent draining of the SFP.

APPLICABLE
SAFETY ANALYSES

The minimum water level in the SFP is an assumption of the fuel handling accident described in the UFSAR (Ref. 3) and Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary as based on this assumption is a small fraction of the 10 CFR 100 (Ref. 5) limits.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Based on the requirements of Reference 4, there must be 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water available, the assumptions of Reference 4 can be used directly. These assumptions include the use of a decontamination factor of 100 in the analysis for iodine. A decontamination factor of 100 enables the analysis to assume that 99% of the total iodine released from the pellet to cladding gap of all dropped fuel assembly rods is retained by the SFP water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.

In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel storage racks, however, there may be < 23 ft of water between the top of the fuel bundle and the surface, indicated by the width of the bundle and difference between the top of the rack and active fuel. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The SFP water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

The SFP water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required during movement of irradiated fuel assemblies within the SFP.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists. Since a fuel handling accident can only occur during movement of fuel, this LCO is not applicable during other conditions. During refueling operations in MODE 6, the SFP water level (and boron concentration) are in equilibrium with the refueling water cavity. The water level under these conditions is then controlled by LCO 3.9.5, "Refueling Cavity Water Level" which requires the refueling cavity water level to be maintained ≥ 23 feet above the top of the reactor vessel flange. A refueling cavity water level of ≥ 23 feet above the top of the reactor vessel flange will result in > 23 feet of water above the top of the active fuel in the storage racks assuming that atmospheric pressure within containment and the Auxiliary Building are equivalent.

ACTIONS

A.1

When the initial conditions assumed in the fuel handling accident analysis cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7:11.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically during movement of irradiated fuel assemblies to ensure the fuel handling accident assumptions are met. The 7 day Frequency is appropriate because the volume in the pool is normally stable and the SFP is designed to prevent drainage below 23 ft. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

Verification of SFP water level can be accomplished by several means. The top of the upper SFP pump suction line is 23 ft above the fuel stored in the pool. If there is ≥ 23 ft of water above the reactor vessel flange (as required by LCO 3.9.5), with equal pressure in the containment and the Auxiliary Building, then at least 23 ft of water is available above the top of the active fuel in the storage racks.

In addition to the physical design features, there are two SFP level alarms (LAL 634) which are available to alert the operators of changing SFP level. A low level alarm will actuate when the SFP water level falls 4 inches or more from the normal level while a high level alarm will actuate when the SFP water level rises 4 inches or more from the normal level. These alarms must receive a calibration consistent with industry practices before they are to be used to meet this SR.

REFERENCES

1. UFSAR, Section 9.1.2.
 2. UFSAR, Section 9.1.3.
 3. UFSAR, Section 15.7.3.
 4. Regulatory Guide 1.25, Rev. 0.
 5. 10 CFR 100.11.
-

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool (SFP) normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both SFP regions is based on the use of unborated water such that configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded.

The double contingency principle discussed in ANSI N-16.1-1975 (Ref. 1) and Reference 2 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.13, "Spent Fuel Pool (SFP) Storage." Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12:1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Ref. 3 and 4). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by voiding (which would result in the addition of negative reactivity) and the SFP geometry which is designed assuming use of unborated water even though soluble boron is available (see Specification 4.3.1.1). The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents which credit use of the soluble boron may be limited to a small fraction of the total operating time.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of the NRC Policy Statement.

LCO

The SFP boron concentration is required to be ≥ 300 ppm. The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 3 and 4 (i.e., a fuel handling accident). This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFP until the fuel assemblies have been verified to be stored correctly.

(continued)

BASES (continued)

APPLICABILITY This LCO applies whenever fuel assemblies are stored in the SFP, until a SFP verification has been performed following the last movement of fuel assemblies in the SFP. The SFP verification is accomplished by performing SR 3.7.13.1 or SR 3.7.13.2 after movement of fuel assemblies depending on which SFP region was affected by the fuel movement. If fuel was moved into both regions, then both SR 3.7.13.1 and SR 3.7.13.2 must be performed after the completion of fuel movement before exiting the Applicability of this LCO. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

This LCO does not apply to fuel movement within a SFP region since the accident analyses assume each region is completely filled in an infinite array.

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to immediately initiate action to perform a SFP verification (SR 3.7.13.1 and SR 3.7.13.2). The performance of this verification removes the plant from the Applicability of this LCO. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.

This SR is required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished (i.e., SR 3.7.13.1 and 3.7.13.2).

REFERENCES

1. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
 2. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
 3. Westinghouse, "Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters," dated June 1994.
 4. UFSAR, Section 15.7.3.
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B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND

The spent fuel pool (SFP) is divided into two separate and distinct regions (see Figure B 3.7.13-1) which, for the purpose of criticality considerations, are considered as separate pools (Ref. 1). Region 1, with 176 storage positions, is designed to accommodate new or spent fuel utilizing a two of four checkerboard arrangement. A fuel assembly with an enrichment of ≤ 4.05 wt% can be stored at any available location in Region 1 since the accident analyses were performed assuming that Region 1 was filled with fuel assemblies of this enrichment. A fuel assembly with an enrichment > 4.05 wt% U-235 can also be stored in Region 1 provided that integral burnable poisons are present in the assemblies such that k -infinity is ≤ 1.458 . The existing design uses Integral Fuel Burnable Absorbers (IFBAs) as the poison for fuel assemblies with enrichments > 4.05 wt%. IFBAs consist of neutron absorbing material which provides equivalencing reactivity holddown (i.e., neutron poison) that allows storage of higher enrichment fuel. The neutron absorbing material is a non-removable or integral part of the fuel assembly once it is applied. The infinite multiplication factor, K -infinity, is a reference criticality point of each fuel assembly that if maintained ≤ 1.458 , will result in a $k_{eff} \leq 0.95$ for Region 1. The K -infinity limit is derived for constant conditions of normal reactor core configuration (i.e., typical geometry of fuel assemblies in vertical position arranged in an infinite array) at cold conditions (i.e., 68°F and 14.7 psia).

Region 2, with 840 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.13-1, in the accompanying LCO. The storage of fuel assemblies which are within the acceptable range of Figure 3.7.13-1 in Region 2 ensures a $K_{eff} \leq 0.95$ in this region.

(continued)



BASES

BACKGROUND
(continued)

Consolidated rod storage canisters can also be stored in either region in the SFP provided that the minimum burnup of Figure 3.7.13-1 is met. In addition, all canisters placed into service after 1994 must have ≤ 144 rods or ≥ 256 rods (Ref. 2). The canisters are stainless steel containers which contain the fuel rods of a maximum of two fuel assemblies (i.e., 358 rods). All bowed, broken, or otherwise failed fuel rods are first stored in a stainless steel tube of 0.75 inch outer diameter before being placed in a canister. Each canister will accommodate 110 failed fuel rod tubes.

The water in the SFP normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water such that configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded.

The double contingency principle discussed in ANSI N16.1-1975 (Ref. 3) and Reference 4 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with this LCO. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 2 and 5). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by voiding (which would result in the addition of negative reactivity) and the SFP geometry which is designed assuming use of unborated water even though soluble boron is available (see Specification 4.3.1.1). The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents which credit use of the soluble boron may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the SFP ensure the k_{eff} of the SFP will always remain < 0.95 , assuming the pool to be flooded with unborated water (Specification 4.3.1.1). For fuel assemblies stored in Region 1, each assembly must have a K-infinity of ≤ 1.458 .

For fuel assemblies stored in Region 2, initial enrichment and burnup shall be within the acceptable area of the Figure 3.7.13-1. The x-axis of Figure 3.7.13-1 is the nominal U-235 enrichment wt% which does not include the ± 0.05 wt% tolerance that is allowed for fuel manufacturing and listed in Specification 4.3.1.1.

(continued)

BASES (continued)

APPLICABILITY This LCO applies whenever any fuel assembly is stored in the SFP.

ACTIONS A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.7.13.1

This SR verifies by administrative means that the K-infinity of each fuel assembly is ≤ 1.458 prior to storage in Region 1. If the initial enrichment of a fuel assembly is ≤ 4.05 wt%, a K-infinity of ≤ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that k-infinity ≤ 1.458 prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.7.13.1 (continued)

Though not required for this LCO, this SR must also be performed after completion of fuel movement into Region 1 to exit the Applicability of LCO 3.7.12, "SFP Boron Concentration."

This SR is modified by a Note which states that this verification is not required when transferring a fuel assembly from Region 2 to Region 1. The verification is not required since Region 2 is the limiting SFP region, and as such, the fuel has already been verified to be acceptable for storage in Region 1.

SR 3.7.13.2

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-1 in the accompanying LCO prior to storage in Region 2. Once a fuel assembly has been verified to be within the acceptable range of Figure 3.7.13-1, further verifications are no longer required since the initial enrichment or burnup will not adversely change. For fuel assemblies in the unacceptable range of Figure 3.7.13-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

Though not required for this LCO, this SR must also be performed after completion of fuel movement into Region 2 to exit the Applicability of LCO 3.7.12.

REFERENCES

1. UFSAR, Section 9.1.2.
2. Westinghouse, "Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters," dated June 1994.
3. ANSI N16.1-1975; "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

(continued)

BASES

REFERENCES
(continued)

4. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
 5. UFSAR, Section 15.7.3.
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Spent Fuel Storage Racks

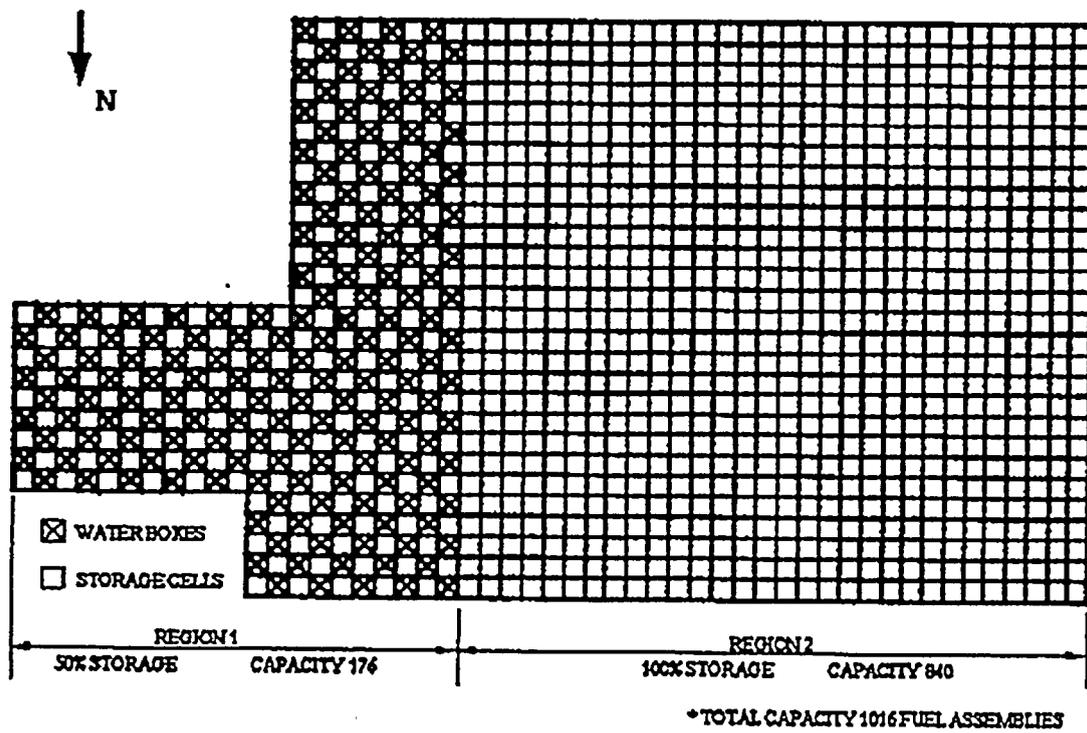


Figure B 3.7.13-1
Spent Fuel Pool

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator (SG) tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes can be observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and Design Basis accidents (DBAs).

This limit is based on an activity value that might be expected from a 0.1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 $\mu\text{Ci/gm}$ (LCO 3.4.16, "RCS Specific Activity"). A steam line break (SLB) is assumed to result in the release of the noble gas and iodine activity contained in the SG inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be approximately 10 rem if the main steam safety valves (MSSVs) were left open for 2 hours following a trip from full power. Operating a plant at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The accident analysis of the SLB, (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an SLB do not exceed a small fraction of the plant EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining SG is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valve (ARV). The Auxiliary Feedwater System supplies the necessary makeup to the SG. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the SG connected to the failed steam line is assumed to be released directly to the environment within 60 seconds. The unaffected SG is assumed to discharge steam and any entrained activity through the MSSVs and ARV for the initial two hours of the event. Primary coolant was assumed to be 3.0 $\mu\text{Ci/gm}$ for this analysis based on previously allowed limits which is a factor of three greater than current limits specified in LCO 3.4.16. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The specific activity of the secondary coolant is required to be $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a DBA to a small fraction of the required limit (Ref. 1).

(continued)

BASES

LCO
(continued)

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the plant in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere from a SLB.

In MODES 5 and 6, the SGs are not being used for heat removal. Both the RCS and SGs are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11..
 2. Letter from D. M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: "SEP Topic, XV-2, Spectrum of Steam System Piping Failures Inside and Outside Containment; XV-12, Spectrum of Rod Ejection Accidents; XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment; XV-17, Steam Generator Tube Failure; and XV-20, Radiological Consequences of Fuel Damaging Accidents - R.E. Ginna," dated September 24, 1981.
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3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - MODES 1, 2, 3, and 4

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

- a. One qualified independent offsite power circuit connected between the offsite transmission network and each of the onsite 480 V safeguards buses required by LCO 3.8.9, "Distribution Subsystems - MODES 1, 2, 3, and 4"; and
- b. Two emergency diesel generators (DGs) capable of supplying their respective onsite 480 V safeguards buses required by LCO 3.8.9.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Offsite power to one or more 480 V safeguards bus(es) inoperable.	A.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition A concurrent with inoperability of redundant required feature(s)
	<u>AND</u> A.2 Restore offsite circuit to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for the offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	<u>OR</u> B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	<u>AND</u> B.4 Restore DG to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Offsite power to one or more 480 V safeguards bus(es) inoperable.</p> <p><u>AND</u></p> <p>One DG inoperable.</p>	<p>-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4," when Condition C is entered with no AC power source to one distribution train. -----</p> <p>C.1 Restore required offsite circuit to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2 Restore DG to OPERABLE status.</p>	<p>12 hours</p> <p>12 hours</p>
<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
<p>E. Two DGs inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for the offsite circuit to each of the 480 V safeguards buses.</p>	<p>7 days</p>
<p>SR 3.8.1.2 -----NOTES----- 1. Performance of SR 3.8.1.9 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ----- Verify each DG starts from standby conditions and achieves rated voltage and frequency.</p>	<p>31 days</p>
<p>SR 3.8.1.3 -----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.9. ----- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes and < 120 minutes at a load ≥ 1950 kW and < 2250 kW.</p>	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.4 Verify the fuel oil level in each day tank.	31 days
SR 3.8.1.5 Verify the DG fuel oil transfer system operates to transfer fuel oil from each storage tank to the associated day tank.	31 days
SR 3.8.1.6 Verify transfer of AC power sources from the 50/50 mode to the 100/0 mode and 0/100 mode.	24 months
SR 3.8.1.7 -----NOTES----- 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. ----- Verify each DG does not trip during and following a load rejection of ≥ 295 kW.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.8 -----NOTES-----</p> <ol style="list-style-type: none"> 1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 2. Credit may be taken for unplanned events that satisfy this SR. <p>-----</p> <p>Verify each DG automatic trips are bypassed on an actual or simulated safety injection (SI) signal except:</p> <ol style="list-style-type: none"> a. Engine overspeed; b. Low lube oil pressure; and c. Start failure (overcrank) relay. 	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. 3. Credit may be taken for unplanned events that satisfy this SR. <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal:</p> <ol style="list-style-type: none"> a. De-energization of 480 V safeguards buses; b. Load shedding from 480 V safeguards buses; and c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. energizes permanently connected loads, 2. energizes auto-connected emergency loads through the load sequencer, and 3. supplies permanently and auto-connected emergency loads for ≥ 5 minutes. 	<p>24 months</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - MODES 5 and 6

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

- a. One qualified independent offsite power circuit connected between the offsite transmission network and each of the onsite 480 V safeguard buses required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6"; and
- b. One emergency diesel generator (DG) capable of supplying one train of the onsite 480 V safeguard bus(es) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Offsite power to one or more required 480 V safeguards bus(es) inoperable.</p>	<p>-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. -----</p> <p>A.1 Declare affected required feature(s) inoperable.</p> <p><u>OR</u></p> <p>A.2.1 Suspend CORE ALTERATIONS.</p> <p><u>AND</u></p> <p>A.2.2 Suspend movement of irradiated fuel assemblies.</p> <p><u>AND</u></p>	<p>Immediately</p> <p>Immediately</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2.3 Initiate action to suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.</p>	<p>Immediately</p> <p>Immediately</p>
B. DG to the required 480 V safeguards bus(es) inoperable.	<p>B.1 Suspend CORE ALTERATIONS.</p> <p><u>AND</u></p> <p>B.2 Suspend movement of irradiated fuel assemblies.</p> <p><u>AND</u></p> <p>B.3 Initiate action to suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>B.4 Initiate action to restore required DG to OPERABLE status.</p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	For AC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.1.1 SR 3.8.1.4 SR 3.8.1.2 SR 3.8.1.5	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil

LCO 3.8.3 The stored diesel fuel oil shall be within limits for each required emergency diesel generator (DG).

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2,
"AC Sources - MODES 5 and 6."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DGs with onsite fuel oil supply not within limit.	A.1 Restore fuel oil level to within limit.	12 hours
B. One or more required DGs with stored fuel oil total particulates not within limit.	B.1 Restore fuel oil total particulates within limit.	7 days
C. One or more DGs with new fuel oil properties not within limits.	C.1 Restore stored fuel oil properties within limits.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more required DGs diesel fuel oil not within limits for reasons other than Condition A, B, or C.</p>	<p>D.1 Declare associated DG inoperable.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.3.1 Verify each fuel oil storage tank contains ≥ 5000 gal of diesel fuel oil for each required DG.</p>	<p>31 days</p>
<p>SR 3.8.3.2 Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.</p>	<p>In accordance with the Diesel Fuel Oil Testing Program</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources—MODES 1, 2, 3, and 4

LCO 3.8.4 The Train A and Train B DC electrical power sources shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power source inoperable.	A.1 Restore DC electrical power source to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in Mode 5.	36 hours
C. Both DC electrical power sources inoperable.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is ≥ 129 V on float charge.	7 days
SR 3.8.4.2 -----NOTES----- 1. SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.3 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -----</p> <p>Verify battery capacity is \geq 80% of the manufacturer's rating when subjected to a performance discharge test.</p>	<p>60 months</p> <p><u>AND</u></p> <p>12 months when battery shows degradation, or has reached 85% of expected life with capacity < 100% of manufacturer's rating</p> <p><u>AND</u></p> <p>24 months when battery has reached 85% of the expected life with capacity \geq 100% of manufacturer's rating</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - MODES 5 and 6

LCO 3.8.5 DC electrical power sources shall be OPERABLE to support the DC electrical power distribution subsystem required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power source(s) inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required DC electrical power source(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.5.1 For DC sources required to be OPERABLE, SR 3.8.4.1 is applicable.	In accordance with applicable SR

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B batteries shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DC electrical power sources are required to be OPERABLE by LCO 3.8.5, "DC Sources—MODES 5 and 6."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within limits.	A.1 Declare associated battery inoperable.	Immediately



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.6.1 Verify electrolyte level of each connected battery cell is above the top of the plates and not overflowing.	31 days
SR 3.8.6.2 Verify the float voltage of each connected battery cell is > 2.07 V.	31 days
SR 3.8.6.3 Verify specific gravity of the designated pilot cell in each battery is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B.	31 days
SR 3.8.6.4 Verify average electrolyte temperature of the designated pilot cell in each battery is $\geq 55^{\circ}\text{F}$.	31 days
SR 3.8.6.5 Verify average electrolyte temperature of every fifth cell of each battery is $\geq 55^{\circ}\text{F}$.	92 days
SR 3.8.6.6 Verify specific gravity of each connected battery cell is: <ul style="list-style-type: none"> a. Not more than 0.020 below average of all connected cells, and b. Average of all connected cells is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. 	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 AC Instrument Bus Sources - MODES 1, 2, 3, and 4

LC0 3.8.7 The following AC instrument bus power sources shall be OPERABLE:

- a. Inverters for Instrument Buses A and C; and
- b. Class 1E constant voltage transformer (CVT) for Instrument Bus B.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One inverter inoperable.	A.1 Power AC instrument bus from its Class 1E or non-Class 1E CVT.	2 hours
	<u>AND</u>	
	A.2 Power AC instrument bus from its Class 1E CVT.	24 hours
	<u>AND</u>	
	A.3 Restore inverter to OPERABLE status.	72 hours
B. Class 1E CVT for AC Instrument Bus B inoperable.	B.1 Power AC Instrument Bus B from its non-Class 1E CVT.	2 hours
	<u>AND</u>	
	B.2 Restore Class 1E CVT for AC Instrument Bus B to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours
D. Two or more required instrument bus sources inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.7.1 Verify correct static switch alignment to Instrument Bus A and C.	7 days.
SR 3.8.7.2 Verify correct Class 1E CVT alignment to Instrument Bus B.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 AC Instrument Bus Sources - MODES 5 and 6

LCO 3.8.8 AC instrument bus power sources shall be OPERABLE to support the onsite Class 1E AC instrument bus electrical power distribution subsystem required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC instrument bus power source(s) inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
	A.2.4 Initiate action to restore required AC instrument bus power source(s) to OPERABLE status.	Immediately



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct static switch alignment to required AC instrument bus(es).	7 days
SR 3.8.8.2	Verify correct Class 1E CVT alignment to the required AC instrument bus.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - MODES 1, 2, 3, and 4

LCO 3.8.9 Train A and Train B of the following electrical power distribution subsystems shall be OPERABLE:

- a. AC power;
- b. AC instrument bus power; and
- c. DC power.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution train inoperable.	A.1 Restore AC electrical power distribution train to OPERABLE status.	8 hours
B. One AC instrument bus electrical power distribution train inoperable.	B.1 Restore AC instrument bus electrical power distribution train to OPERABLE status.	2 hours
C. One DC electrical power distribution train inoperable.	C.1 Restore DC electrical power distribution train to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Conditions A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 5.	36 hours
E. Two trains with inoperable electrical power distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.9.1 Verify correct breaker alignments and voltage to required electrical power trains.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - MODES 5 and 6

LCO 3.8.10 The necessary trains(s) of the following electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE:

- a. AC power;
- b. AC instrument bus power; and
- c. DC power.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required electrical power distribution train(s) inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
		(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential active components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for each of these electrical power sources which are further divided and organized based on voltage considerations and safety classification. This LCO is provided to specify the minimum sources of AC power which are required to supply the 480 V safeguards buses and associated distribution subsystem during MODES 1, 2, 3, and 4.

The plant AC sources consist of an independent offsite power source and the onsite standby emergency power source (Ref. 1). Atomic Industrial Forum (AIF) GDC 39 (Ref. 2) requires emergency power sources be provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning of the Engineered Safety Features (ESF) and protection systems. The offsite and onsite AC sources can each supply power to 480 V safeguards buses to ensure that reliable power is available during any normal or emergency mode of plant operation. The 480 V safeguards buses are divided into redundant trains so that the loss of any one train does not prevent the minimum safety functions from being performed. Safeguards Buses 14 and 18 are associated with Train A and safeguards Buses 16 and 17 are associated with Train B. Since only the onsite standby power source is classified as Class 1E, the offsite power source is not required to be separated into redundant trains.

(continued)

BASES

BACKGROUND
(continued)

The independent offsite power source consists of breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite 480 V safeguards buses. The independent offsite power source essentially begins from two station auxiliary transformers (SAT 12A and 12B) each supplied from an independent transmission line emanating from separate switchyards (see Figure B 3.8.1-1). SAT 12A is connected to the 34.5 kV transmission system (circuit 751) and SAT 12B is connected to the plant 115 kV switchyard (circuit 767). The SATs may be configured in the following modes:

- a. SAT 12A (or SAT 12B) supplies safeguards Buses 16 and 17 and SAT 12B (or SAT 12A) supplies safeguards Buses 14 and 18 (50/50 mode);
- b. SAT 12A supplies all safeguards Buses (0/100 mode); or
- c. SAT 12B supplies all safeguards Buses (100/0 mode).

The preferred configuration is the 50/50 mode; however, all three modes of operation meet applicable design requirements for normal operation (Ref. 1). Offsite power can also be provided during an emergency through the plant auxiliary transformer 11 by backfeeding from the 115 kV transmission system and main transformer.

SATs 12A and 12B are each connected to two non-Class 1E, 4.16 kV buses (12A and 12B). The 4.16 kV Bus 12A feeds the Class 1E loads on the 480 V safeguards Buses 14 and 18 and 4.16 kV Bus 12B feeds the Class 1E loads on the 480 V safeguards Buses 16 and 17 (see Figure B 3.8.1-1). Loss of power to any of the safeguards buses, as a result of inoperable offsite circuit component(s), is a loss of offsite power. The offsite power source ends after the feeder breaker supplying each 480 V safeguards bus.

(continued)

BASES

BACKGROUND
(continued)

The onsite standby power sources consist of two 1950 kW continuous rating emergency diesel generators (DGs) connected to the safeguards buses to supply emergency power in the event of loss of all other AC power. The DGs are located in separate rooms in a Seismic Category I structure located adjacent to the northeast wall of the Turbine Building. Each DG room has its own ventilation system. The ventilation system is designed to maintain the DG room between 60°F and 104°F and to remove any hydrocarbon gases in the room (Ref. 3). Each ventilation system consists of two fans and associated ductwork and dampers that fail open on loss of instrument air and control power. One fan is designed to start on DG actuation with a second fan designed to start when the room temperature reaches 90°F. The second fan's discharge air flow is directed to the DG control panel and has a delayed start to prevent potentially freezing the cooling water jacket piping during cold weather conditions.

The DGs utilize an air motor for starting. The air motor is supplied by two receivers which provide sufficient air for five DG starts before requiring a recharge of the receivers. The DGs are supplied by separate fuel oil day tanks which can be cross-tied if required. Additional fuel oil can be transferred from redundant underground fuel oil storage tanks. A dedicated fuel oil transfer pump is used for this transfer. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank, to result in the loss of more than one DG.

DG A is dedicated to safeguards Buses 14 and 18 and DG B is dedicated to safeguards Buses 16 and 17. A DG starts automatically on a safety injection (SI) signal or on an undervoltage signal on its corresponding 480 V buses (refer to LCO 3.3.4, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). In the event of only an SI signal, the DGs automatically start and operate in the standby mode without tying to the safeguards buses.

(continued)

BASES

BACKGROUND
(continued)

In the event of loss of offsite power, or abnormal offsite power where offsite power is tripped as a consequence of bus undervoltage or degraded voltage, the DGs automatically start and tie to their respective buses. All bus loads except for the containment spray (CS) pump, component cooling water (CCW) pump and safety related motor control centers are tripped upon actuation of the undervoltage relays. This is independent of or coincident with an SI signal. Once the undervoltage relay resets independent of a SI signal, the operator may manually connect loads onto the bus(es). During a coincident SI signal, the CCW pump is also tripped and loads are sequentially connected to their respective buses by the automatic load sequencer.

In the event of loss of offsite power to only one safeguards bus in a train, the DG will automatically start and tie only to the affected bus. During a coincident SI signal, the normal feed breaker on the second bus on the affected train will be tripped by the undervoltage relay on the failed bus causing the DG to automatically tie to both buses. This condition will then actuate the automatic load sequencer.

In the event of a loss of offsite power and a coincident SI signal, the electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA). Certain required plant loads are returned to service in a predetermined sequence by the automatic load sequencer in order to prevent overloading the DG during the start process. Within approximately 1 minute after the initiating signal is received, all loads needed to recover the plant or maintain it in a safe condition are returned to service.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The initial conditions of DBA and transient analyses (Refs. 4 and 5) assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This results in maintaining at least one train of the onsite standby power or offsite AC sources OPERABLE in the event of:

- a. An assumed loss of all AC offsite power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC electrical power sources ensures that one train of offsite or onsite standby AC power is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one train of onsite standby power).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of offsite power also ensures that at least one AC power source is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 1). Therefore, the requirements of GDC 17 (Ref. 6) can be met at all times.

The DGs are designed to operate following a DBA or anticipated operational occurrence (AOO) until offsite power can be restored. An AOO is defined as a Condition 2 event in Reference 7 (i.e., events which can be expected to occur during a calendar year with moderate frequency). The DGs are required to start within 10 seconds and begin loading. The DGs can begin receiving up to 30% of design loads after the 10 second start time and can accept 100% of design loads within 30 seconds. The DGs are manually loaded if only an undervoltage signal is present and load sequenced if a coincident undervoltage and SI signal is present. The loads are sequenced as follows (assume SI signal at 0 seconds):

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	DG Load	DG A Time	DG B Time
	480V safeguards buses and CS pumps	10	10
	SI pump A and B	15	15
	SI pump C	20	22
	Residual heat removal pump	25	27
	Selected service water pump	30	32
	First containment recirculation fan cooler	35	37
	Second containment recirculation fan cooler	40	42
	Motor driven auxiliary feedwater pump	45	47

Since the DGs must start and begin loading within 10 seconds, only one air start must be available in the air receivers as assumed in the accident analyses. The long term operation of the DGs (until offsite power is restored) is discussed in LCO 3.8.3, "Diesel Fuel Oil."

The AC sources satisfy Criterion 3 of NRC Policy Statement.

LCO

One qualified independent offsite power circuit connected between the offsite transmission network and the onsite 480 V safeguards buses and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA.

An OPERABLE qualified independent offsite power circuit is one that is capable of maintaining rated voltage, and accepting required loads during an accident, while connected to the 480 V safeguards buses required by LCO 3.8.9, "Distribution Subsystems - MODES 1, 2, 3, and 4." Power from either offsite power circuit 751 or 767 satisfies this requirement.

A DG is considered OPERABLE when:

- a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;

(continued)

BASES

LCO
(continued)

- b. All loads on each 480 V safeguards bus except for the safety related motor control centers, CCW pump, and CS pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident SI and undervoltage signal);
- c. The DG is capable of accepting required loads both manually and within the assumed loading sequence intervals following a coincident SI and undervoltage signal, and continue to operate until offsite power can be restored to the safeguards bus (i.e., 40 hours);
- d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;
- e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and
- f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

The AC sources in one train must be separate and independent of the AC sources in the other train. For the DGs, separation and independence must be complete assuming a single active failure. For the independent offsite power source, separation and independence are to the extent practical (i.e., operation is preferred in the 50/50 mode, but may also exist in the 100/0 or 0/100 mode).

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

(continued)



BASES

APPLICABILITY
(continued)

- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - MODES 5 and 6."

ACTIONS

A.1 and A.2

With offsite power to one or more 480V safeguard bus(es) inoperable, assurance must be provided that a coincident single failure will not result in a complete loss of required safety features. If the redundant safety feature to the component or train affected by the loss of offsite power is also unavailable, the assumption that two complete safety trains are OPERABLE may no longer exist. As an example, if offsite power were unavailable to 480 V Bus 14, DG A could supply the necessary power to the bus. If residual heat removal pump (RHR) B (supplied power by Bus 16) were inoperable at the same time, or at any time after the loss of offsite power to Bus 14, a loss of redundant required safety features exists since a failure of DG A would result in the loss of emergency core cooling. Therefore, RHR pump A on Bus 14 would have to be declared inoperable within 12 hours after RHR pump B and offsite power to Bus 14 were declared unavailable.

The Completion Time of 12 hours as provided by Required Action A.1 to declare the required safety features inoperable is based on the fact that it is less than the Completion Time for restoring OPERABILITY of the offsite power circuit and all safety features affected by the loss of the 480 V bus. A shorter Completion Time is provided since the required safety features have been potentially degraded by the loss of offsite power (i.e., using the same example as above, the 72 hour Completion Time for restoring RHR pump B was developed assuming that RHR pump A had both offsite and onsite standby emergency power available). Therefore, a penalty is assessed to only allow 12 hours in this configuration.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time for Required Action A.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time is an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that:

- a. There is no offsite power available to one or more 480 V safeguards bus; and
- b. A redundant required feature is inoperable on a second 480 V safeguards bus.

If at any time during the existence of Condition A, a redundant required feature becomes inoperable, this Completion Time begins to be tracked. Required Action A.1 can be exited if the inoperable DG or the required feature on the OPERABLE DG is restored to OPERABLE status.

The level of degradation during Condition A means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite standby AC sources have not been degraded. This level of degradation generally corresponds to either:

- a. Loss of offsite power sources to SAT 12A and/or SAT 12B;
- b. Failure of SAT 12A or 12B or 4.16 kV Bus 12A or 12B;
or
- c. Failure of a station service transformer supplying a 480 V safeguards bus.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

With a total loss of the offsite power sources to SAT 12A and 12B, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident for either train. With loss of offsite power to SAT 12A or 12B, failure of SAT 12A or 12B, or failure of Bus 12A or 12B, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident for a single AC electrical train. With a failure of a station service transformer, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the consequences of an accident for one 480 V safeguards bus in one AC electrical train. In all cases, sufficient onsite AC sources are available to maintain the plant in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 72 hour Completion Time provides a period of time to effect restoration of the offsite circuit commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

B.1

With one DG inoperable, it is necessary to verify the availability of the offsite circuit to each of the affected 480 V safeguards buses on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met (i.e., Condition D would not apply). However, if a circuit fails to pass SR 3.8.1.1, it is inoperable and Condition C would be entered. The Completion Time of 1 hour to perform SR 3.8.1.1 is based on the importance of this verification to ensure that offsite power is available to the affected bus. The Frequency of once per 8 hours thereafter is based on the alarms and indications of breaker status that are available in the control room.

(continued)



BASES

ACTIONS
(continued)

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of a safety feature. These features are designed with redundant safety related trains which are supplied power from separate and independent on-site power sources. If one on-site power source is inoperable, it must be assured that the redundant safety related train supplied by the OPERABLE DG is available to provide the necessary safety function.

The Completion Time of 4 hours for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time is an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. An inoperable DG exists; and
- b. A required feature on the other train (Train A or Train B) is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature supported by the OPERABLE DG subsequently becomes inoperable, this Completion Time would begin to be tracked. Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are supplied power by the OPERABLE DG, results in starting the Completion Time for Required Action B.2. In this Condition, the remaining OPERABLE DG and the offsite circuit are adequate to supply electrical power to the on-site 480 V safeguards buses.

(continued)

BASES

ACTIONS

B.2 (continued)

The Completion Time of 4 hours to declare the required safety features inoperable is based on the fact that it is less than the Completion Time for restoring OPERABILITY of the DG and all safety features supported by the DG. A shorter Completion Time is provided since the required safety features have been potentially degraded by the inoperable DG. Therefore, a penalty is assessed to only allow 4 hours in this configuration. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Required Action B.2 can be exited if the inoperable DG or the required feature on the OPERABLE DG is restored to OPERABLE status.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined within 24 hours that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 is not required to be performed. If the cause of inoperability is determined to exist on the other DG, the second DG would be declared inoperable upon discovery and Condition E would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the second DG within 24 hours, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, activities must continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

The 24 hour Completion Time is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG (Ref. 8).

(continued)

BASES

ACTIONS
(continued)

B.4

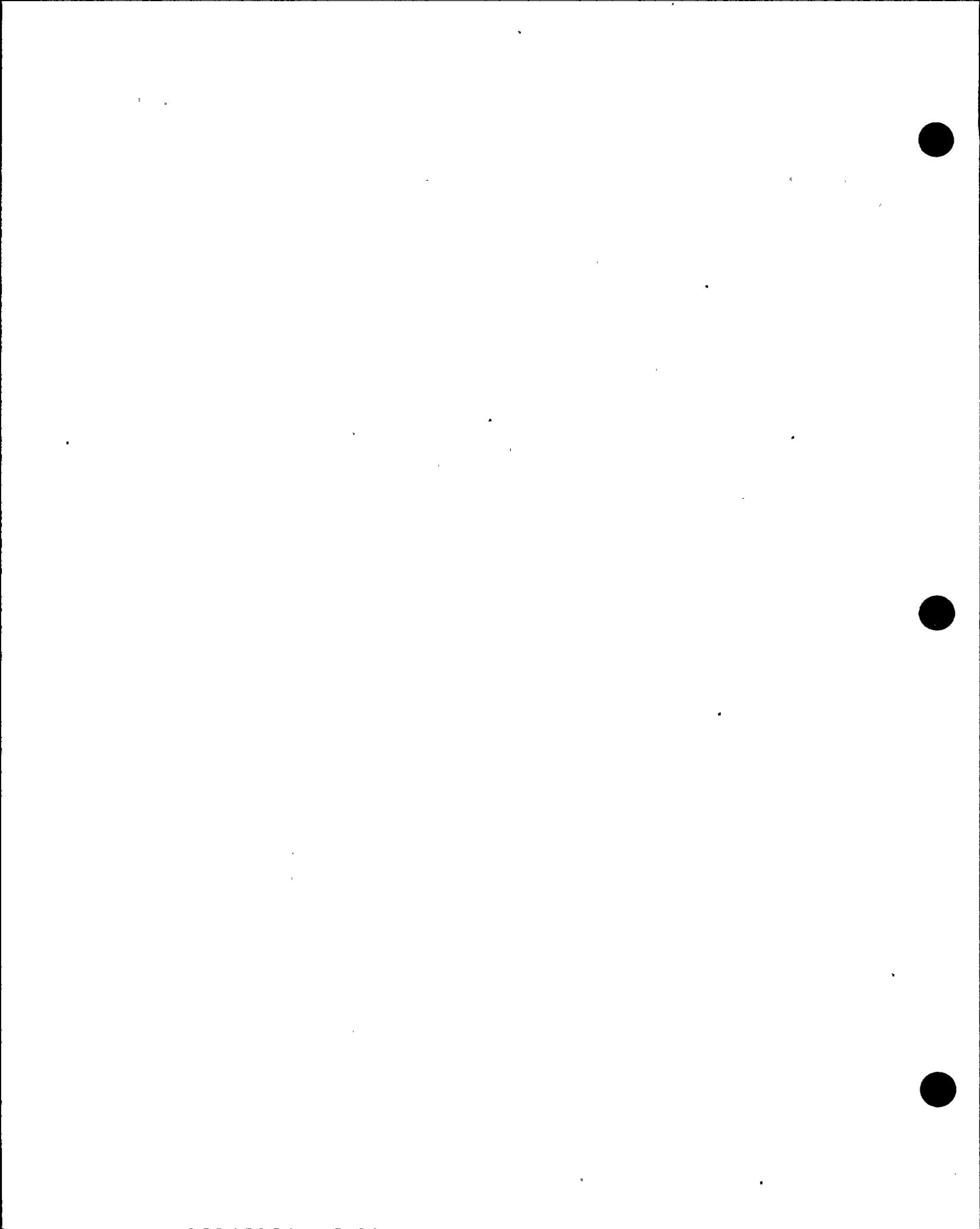
With one inoperable DG, the remaining OPERABLE DG and the offsite circuit are adequate to supply electrical power to the onsite 480 V safeguards buses. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

C.1

With offsite power to one or more 480 V safeguards bus(es) and one DG inoperable, redundancy is lost in both the offsite and onsite AC electrical power systems. Since power system redundancy is provided by these two diverse sources of power, the AC power sources are only degraded and no loss of safety function has occurred since at least one DG and potentially one offsite AC power source are available. However, the plant is vulnerable to a single failure which could result in the loss of multiple safety functions. Therefore, a Completion Time of 12 hours is provided to either restore the offsite power circuit or the DG to OPERABLE status. This Completion Time is less than that for an inoperable offsite power source or an inoperable DG due to the single failure vulnerability of this configuration.

As discussed in LCO 3.0.6, the AC electrical power distribution subsystem ACTIONS would not be entered even if all AC sources to either train were inoperable, resulting in de-energization. Therefore, the Required Actions of this Condition are modified by a Note which states that the Required Actions of LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4" must also be immediately entered with no AC power source to one distribution train. This allows Condition C to provide requirements for the loss of an offsite power circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

(continued)



BASES

ACTIONS
(continued)

D.1 and D.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If both DGs are inoperable, a loss of safety function would exist if offsite power were unavailable; therefore, LCO 3.0.3 must be entered.

SURVEILLANCE
REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function (Ref. 2). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions).

SR 3.8.1.1

This SR ensures proper circuit continuity for the independent offsite power source to each of the onsite 480 V safeguards buses and availability of offsite AC electrical power. Checking breaker alignment and indicated power availability verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their qualified power source. The Frequency of 7 days is adequate since breaker position is not likely to change without the operators knowledge and because alarms and indications of breaker status are available in the control room.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.2

This SR verifies that each DG starts from standby conditions and achieves rated voltage and frequency. This ensures the availability of the DG to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition. The DG voltage control may be either in manual or automatic during the performance of this SR. The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by two Notes. Note 1 indicates that performance of SR 3.8.1.9 satisfies this SR since SR 3.8.1.9 is a complete test of the DG. The second Note states that all DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes the wear on moving parts that do not get lubricated when the engine is not running.

SR 3.8.1.3

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures. A maximum run time not to exceed 120 minutes minimizes the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.85 lagging and 0.95 lagging. The upper load band limit of 2250 kW is provided to avoid routine overloading of the DG which may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The lower load band limit is the expected maximum load following a DBA.

In addition to verifying the DG capability for synchronizing with the offsite electrical system and accepting loads, the DG ventilation system should also be verified during this surveillance.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.3

The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients outside the load band (e.g., due to changing bus loads), do not invalidate this test. Similarly, momentary power factor transients above or below the administrative limit do not invalidate the test. Note 3 indicates that this Surveillance shall be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful performance of SR 3.8.1.2 or SR 3.8.1.9 must precede this surveillance to prevent unnecessary starts of the DGs.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in each day tank is at or above the level at which fuel oil is automatically added when the fuel oil transfer pump is in auto and the DG is operating. This level ensures adequate fuel oil for a minimum of 1 hour of DG operation at 110% of full load. This is equivalent to a day tank level of 8.25 inches above the tank suction line.

The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.5

This SR demonstrates that each DG fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of the DGs. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic or manual fuel transfer systems are OPERABLE.

The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, since the design of the fuel oil transfer system is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG operation.

SR 3.8.1.6

This SR involves the transfer of the 480 V safeguards bus power supply from the 50/50 mode to the 100/0 mode and 0/100 mode which demonstrates the OPERABILITY of the alternate circuit distribution network to power the required loads. The Frequency of 24 months is based on engineering judgment, taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.7

This SR verifies that each DG does not trip during and following a load rejection of ≥ 295 kW. Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This SR demonstrates the DG load response characteristics and capability to reject the largest single load on the buses supplied by the DG (i.e., a safety injection pump).

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ 0.9 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.8

This SR demonstrates that DG noncritical protective functions (e.g., overcurrent, reverse power, local stop pushbutton) are bypassed on an actual or simulated SI actuation signal, and critical protective functions (engine overspeed, low lube oil pressure, and start failure (overcrank) relay) trip the DG to avert substantial damage to the DG. The noncritical trips are bypassed during DBAs but still provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The Frequency of 24 months is based on engineering judgment, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that performing the Surveillance would remove a required DG from service which is undesirable in these MODES. The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.9

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.9 (continued)

This SR demonstrates the DG operation during an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by three Notes. Note 1 states that all DG starts may be preceded by an engine prelube period which is intended to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine lube oil continuously circulated and temperature maintained consistent with manufacturer recommendations for the DGs. Note 2 states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4 since performing the Surveillance during these MODES would remove a required offsite circuit from service, cause perturbations to the electrical distribution systems, and challenge safety systems. Note 3 acknowledges that credit may be taken for unplanned events that satisfy this SR.

(continued)



BASES (continued)

REFERENCES

1. UFSAR, Chapter 8.
 2. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
 3. UFSAR, Section 9.4.9.5.
 4. UFSAR, Chapter 6.
 5. UFSAR, Chapter 15.
 6. 10 CFR 50, Appendix A, GDC 17.
 7. "American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 8. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
-

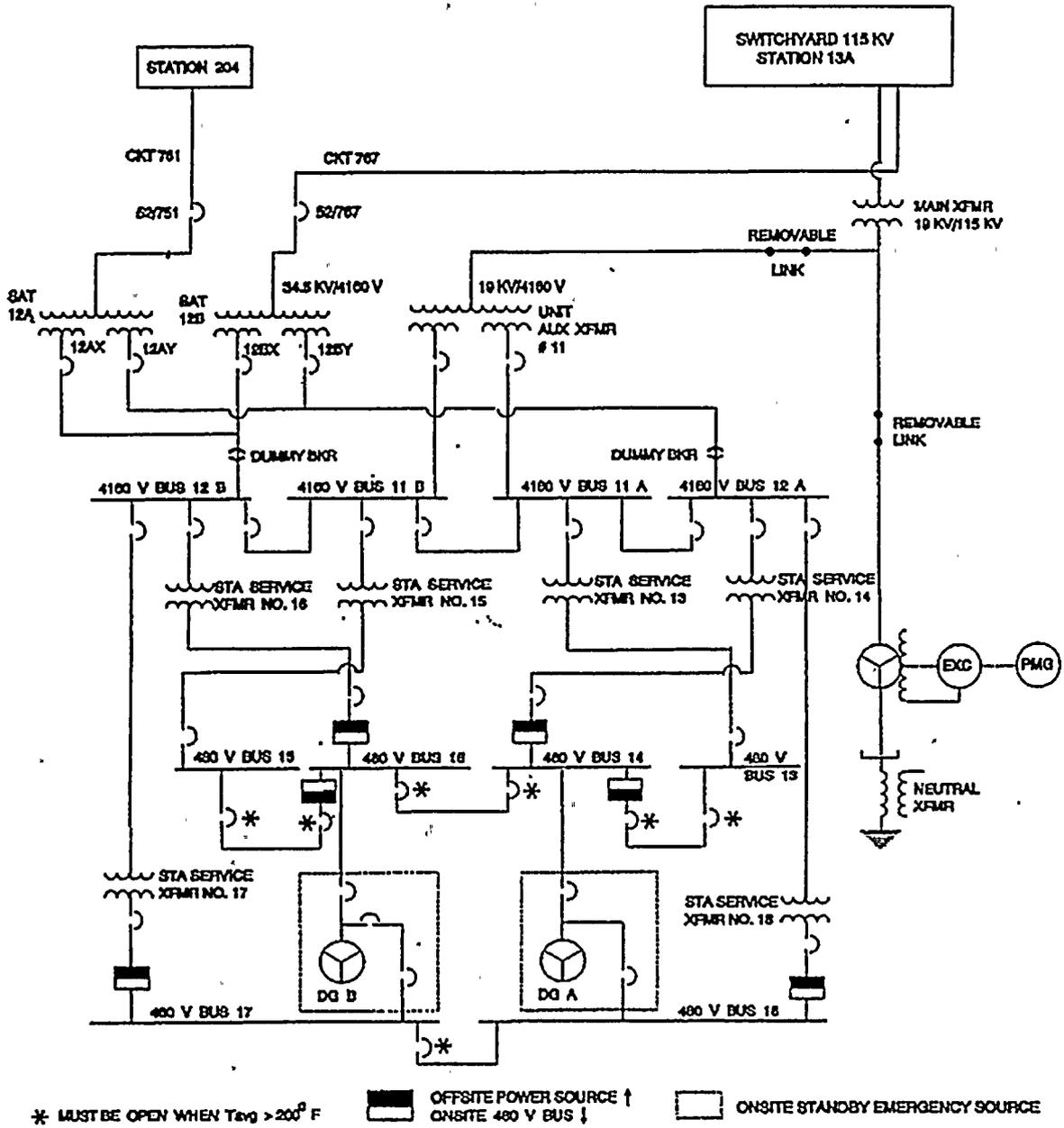


Figure B 3.8.1-1



B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section for Bases 3.8.1, "AC Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6 the minimum required AC sources may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the AC power sources, must be removed from service. The minimum required AC sources is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum AC electrical power sources during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available; and
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available;

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC electrical power sources ensures that one train of the onsite power or offsite AC sources are OPERABLE in the event of:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6 this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One qualified independent offsite power circuit supplying the associated AC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," ensures that all required loads are powered from offsite power. An OPERABLE DG, capable of supporting the distribution system required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the independent offsite power circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

An OPERABLE qualified offsite circuit is one that is capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the 480 V safeguards bus(es). Power from either offsite power circuit 751 or 767, or by backfeeding through auxiliary transformer 11 satisfies this requirement.

(continued)

BASES

LCO
(continued)

A DG is considered OPERABLE when:

- a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;
- b. All loads on each 480 V safeguards bus except for the safety related motor control centers, component cooling water (CCW) pump, and containment spray (CS) pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident safety injection (SI) and undervoltage signal);
- c. The DG is capable of accepting required loads manually. Since most equipment which receives a SI signal are isolated in these MODES due to maintenance or low temperature overpressure protection concerns, and the DBA of concern (i.e., a fuel handling accident) would not generate a SI signal, manual loading of the DGs will most likely be required. These loads must be capable of being added to the OPERABLE DG within 10 minutes;
- d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;
- e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and
- f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

(continued)

BASES (continued)

APPLICABILITY The AC sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of postulated events and to maintain the plant in the cold shutdown or refueling condition are available.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4."

ACTIONS

A.1

As discussed in LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no onsite or offsite AC power to any required 480 V safeguards bus, the ACTIONS for LCO 3.8.10 must also be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite power circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a completely de-energized train.

With offsite power available to one or more required 480 V safeguards bus(es) inoperable, assurance must be provided that there is not a complete loss of required safety features. Although two trains may be required by LCO 3.8.10, one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, or operations involving positive reactivity additions. By allowing the option to declare required features inoperable that are not powered from offsite power, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

(continued)



BASES

ACTIONS
(continued)

A.2.1, A.2.2, A.2.3, and A.2.4

With the offsite power circuit not available to all required AC electrical trains, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

It is further required to immediately initiate action to restore the required offsite power AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required offsite power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

B.1, B.2, B.3, and B.4

With the required DG inoperable, the minimum required diversity of AC power sources is not available. Therefore, it is required that CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions be immediately suspended. Performance of Required Action B.1, B.2, and B.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of temperature control within established procedures.

(continued)



BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

It is further required to immediately initiate action to restore the required DG to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary AC power redundancy to plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DG should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient redundant power.

SURVEILLANCE
REQUIREMENTS

SR 3.8.2.1

This SR requires the performance of SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6.

This SR precludes requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, precludes de-energizing a required 480 V safeguards bus, and precludes unnecessary transfers of the offsite power source configurations. With limited AC sources available, a single event could compromise both the required circuit and the DG. Therefore, the requirement to perform SR 3.8.1.3, and SR 3.8.1.6 through 3.8.1.9 is suspended. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil

BASES

BACKGROUND

Fuel oil is provided to each emergency diesel generator (DG) by a dedicated 350 gal day tank located near the DG. Each day tank is supplied from an associated 6000 gal underground fuel oil storage tank. Each storage tank provides a minimum fuel oil capacity of 5000 gal. The two storage tanks are sufficient to operate both DGs at design ratings for 24 hours. The total minimum fuel oil capacity also ensures that both DGs can operate for a period of 40 hours while providing for a maximum post loss of coolant accident (LOCA) load demand. The maximum load demand is calculated using the assumption that both DGs are available and is less than the DG design rating. The minimum onsite fuel capacity is sufficient to operate the DGs for longer than 8 hours which is the time required to replenish the onsite supply from outside sources (Ref. 1).

Fuel oil is transferred from each storage tank to the associated day tank by a dedicated fuel oil transfer pump. Each fuel oil transfer pump is powered by a 480 V safeguards bus that is backed by the associated DG. One fuel oil transfer pump has the capability to supply both DGs operating with 110% of their design loads. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG.

All outside tanks, pumps, and piping are located underground to protect them from potential missiles. Heat tracing is provided in the exposed suction piping to the fuel oil pumps in the event that heating is lost in the DG rooms. The heat tracing is thermostatically controlled to maintain the fuel oil in the pipe $> 40^{\circ}\text{F}$ which is above the cloud point temperature of the fuel oil (0°F).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 2 and 3), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

Since diesel fuel oil supports the operation of the standby AC power sources, it satisfies Criterion 3 of the NRC Policy Statement.

LCO

Stored onsite diesel fuel oil is required to have sufficient supply for 40 hours of maximum post-LOCA load demand. It is also required to meet specific standards for quality. This requirement, in conjunction with an ability to obtain replacement fuel oil supplies within 8 hours, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4," and LCO 3.8.2, "AC Sources - MODES 5 and 6."

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil supports LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil is required to be within limits in MODES 1, 2, 3 and 4, and when the associated DG is required to be OPERABLE in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1

With one or more required DGs with an onsite supply of < 5000 gal of fuel oil, the assumed 40 hour fuel oil supply for a DG is not available. This circumstance may be caused by events, such as full load operation after an inadvertent start with an initial minimum required fuel oil level, or feed and bleed operations, which may be necessitated by increasing fuel oil particulate levels or any number of other oil quality degradations. Required Action A.1 allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. The Completion Time of 12 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity, the fact that actions will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

If one or more DGs has stored fuel oil with total particulates not within limits for reasons not related to new fuel oil, the fuel oil must be restored within limits within 7 days. The fuel oil particulate properties are verified by SR 3.8.3.2. Trending of particulate levels normally allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample practices (bottom sampling), contaminated sampling equipment, or errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

(continued)



BASES

ACTIONS
(continued)

C.1

With the new fuel oil properties defined in SR 3.8.3.2 not within required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

D.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil not within limits for reasons other than addressed by Conditions A, B, or C (e.g., cloud point temperature reached), the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.1

This SR verifies an onsite supply of ≥ 5000 gal of fuel oil is available for each required DG. This ensures that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 40 hours while providing maximum post-LOCA loads. The 40 hour period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since indications are available to ensure that operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.3.2

This SR provides a means of determining whether new and stored fuel oil has been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. This ensures the availability of high quality fuel oil for the DGs. Fuel oil degradation during long term storage is indicated by an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which could eventually cause engine failure.

A fuel oil sample is analyzed to establish that properties specified in Table 1 of ASTM D975-78 (Ref. 4) for viscosity, water, and sediment are met for the stored fuel oil.

The Frequency of this SR takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals. The Frequency, as specified in the Diesel Fuel Oil Testing Program, is 92 days.

REFERENCES

1. UFSAR, Section 9.5.4.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. ASTM Standards, D975-78, Table 1.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential active components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for these two electrical power sources which are further divided and organized based on voltage considerations and whether they are Class 1E (i.e., supply safety related or engineered safeguards functions) or nonessential. This LCO is provided to specify the minimum sources of DC power which are required to supply the DC buses and their associated distribution system during MODES 1, 2, 3, and 4.

The station DC electrical power subsystem provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC instrument bus power (via inverters). Atomic Industrial Forum (AIF) GDC 39 (Ref. 1) requires emergency power sources be provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems.

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power distribution train (Train A and Train B). Each subsystem consists of one 125 VDC battery, two battery chargers supplied from the 480 V system, distribution panels and buses, and all the associated control equipment and interconnecting cabling (see Figure B 3.8.4-1). The batteries and battery chargers are addressed by this LCO.

(continued)

BASES

BACKGROUND
(continued)

Each battery provides a separate source of DC power independent of AC power. Each of the two batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 105 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 2).

There are four battery chargers available to the batteries. Chargers 1A and 1B are rated at 150 amps and chargers 1A1 and 1B1 are rated at 200 amps. Battery chargers 1A and 1A1 are normally aligned to battery A, and battery chargers 1B and 1B1 are normally aligned to battery B. A charging capacity of at least 150 amps is normally required to supply the necessary DC loads on each train and to provide a full battery charge to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated Design Basis Accident (DBA). The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4," and LCO 3.8.10, "Distribution Systems—MODES 5 and 6."

The DC electrical power distribution subsystem also provide DC electrical power to the inverters, which in turn power the AC instrument buses. The inverters are described in more detail in Bases for LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4," and LCO 3.8.8, "AC Instrument Bus Sources—MODES 5 and 6."

Train A Engineered Safety Feature (ESF) equipment is supplied from battery A, while Train B ESF equipment is supplied from battery B. Additionally, the 480 V ESF switchgear and diesel generator (DG) control panels are supplied from either battery by means of an automatic transfer circuit in the switchgear and control panels. The normal supply from Train A (Buses 14 and 18 and DG A) is from DC distribution panels A. These panels also provide the emergency DC supply for Train B. Similarly, the normal supply from Train B (Buses 16 and 17 and DG B) is from DC distribution panels B. These panels also provide the emergency dc supply for Train A.

(continued)

BASES

BACKGROUND
(continued)

Each 125 VDC battery and associated battery chargers are separately housed in a ventilated room with its associated distribution center. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. The two battery rooms are supplied with ventilation by a common AC powered air conditioning and heating unit which also provides sufficient air changes to prevent hydrogen buildup. A redundant DC powered fan is also available in the event that all AC power is lost. The failure of both the AC powered and DC powered units does not result in unacceptable room service conditions until after 5 hours of continuous battery operation during a DBA (Ref. 2).

The batteries for Train A and Train B DC electrical power distribution subsystem are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity for aging considerations. The minimum voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery.

Each battery charger for the Train A and Train B DC electrical power distribution subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 2).

APPLICABLE
SAFETY ANALYSES

The initial conditions of a DBA and transient analyses (Refs. 3, 4, and 5), assume that ESF systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one train of DC source OPERABLE in the event of:

- a. An assumed loss of all offsite AC power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the DC electrical power sources ensures that one train of DC electrical power is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one DC electrical power source).

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of DC power ensures that at least one DC power source is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 6). Therefore, the requirements of GDC 17 (Ref. 7) can be met at all times.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The Train A and Train B DC electrical power sources, each consisting of one battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Loss of any one train DC electrical power source does not prevent the minimum safety function from being performed.

An OPERABLE DC electrical power source requires the battery and at least one battery charger with a capacity \geq 150 amps to be operating and connected to the associated DC bus. The AC powered and DC powered fan units are not required to be OPERABLE for this LCO, but some form of ventilation may be required for SR 3.8.6.4 and SR 3.8.6.5.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe plant operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in LCO 3.8.5, "DC Sources - MODES 5 and 6."

(continued)

BASES (continued)

ACTIONS

A.1

With one DC electrical power source inoperable, OPERABILITY must be restored within 2 hours. In this Condition, redundancy is lost and only one train is capable to completely respond to an event. If one of the required DC electrical power sources is inoperable, the remaining DC electrical power source has the capacity to support a safe shutdown and to mitigate an accident condition. A subsequent worst case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power distribution subsystem with attendant loss of ESF functions. The 2 hour Completion Time reflects a reasonable time to assess plant status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power source is not restored to OPERABLE status, to prepare to effect an orderly and safe plant shutdown.

B.1 and B.2

If the inoperable DC electrical power source cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both DC electrical power sources are inoperable, a loss of multiple safety functions exists; therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.4.2

This SR verifies that the capacity of each battery is adequate to supply and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test. A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements specified in Reference 2.

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 24 months.

This SR is modified by two Notes. Note 1 states that SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. This substitution is acceptable because SR 3.8.4.3 represents a more severe test of battery capacity than does SR 3.8.4.2. Note 2 states that this surveillance shall not be performed in MODE 1, 2, 3, or 4 because performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.4.3

This Surveillance verifies that each battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test. A battery performance test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity as determined by specified acceptance criteria. The test is intended to determine overall battery degradation due to age and usage.

A battery should be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Frequency for this SR is 60 months when the battery is $< 85\%$ of its expected life with no degradation and 12 months if the battery shows degradation or has reached 85% of its expected life with a capacity $< 100\%$ of the manufacturer's rating. When the battery has reached 85% of its expected life with capacity $\geq 100\%$ of the manufacturer's rating, the Frequency becomes 24 months. Battery degradation is indicated when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is $\geq 10\%$ below the manufacturer rating. These Frequencies are considered acceptable based on the testing being performed in a conservative manner relative to the battery life and degradation. This ensures that battery capacity is adequately monitored and that the battery remains capable of performing its intended function.

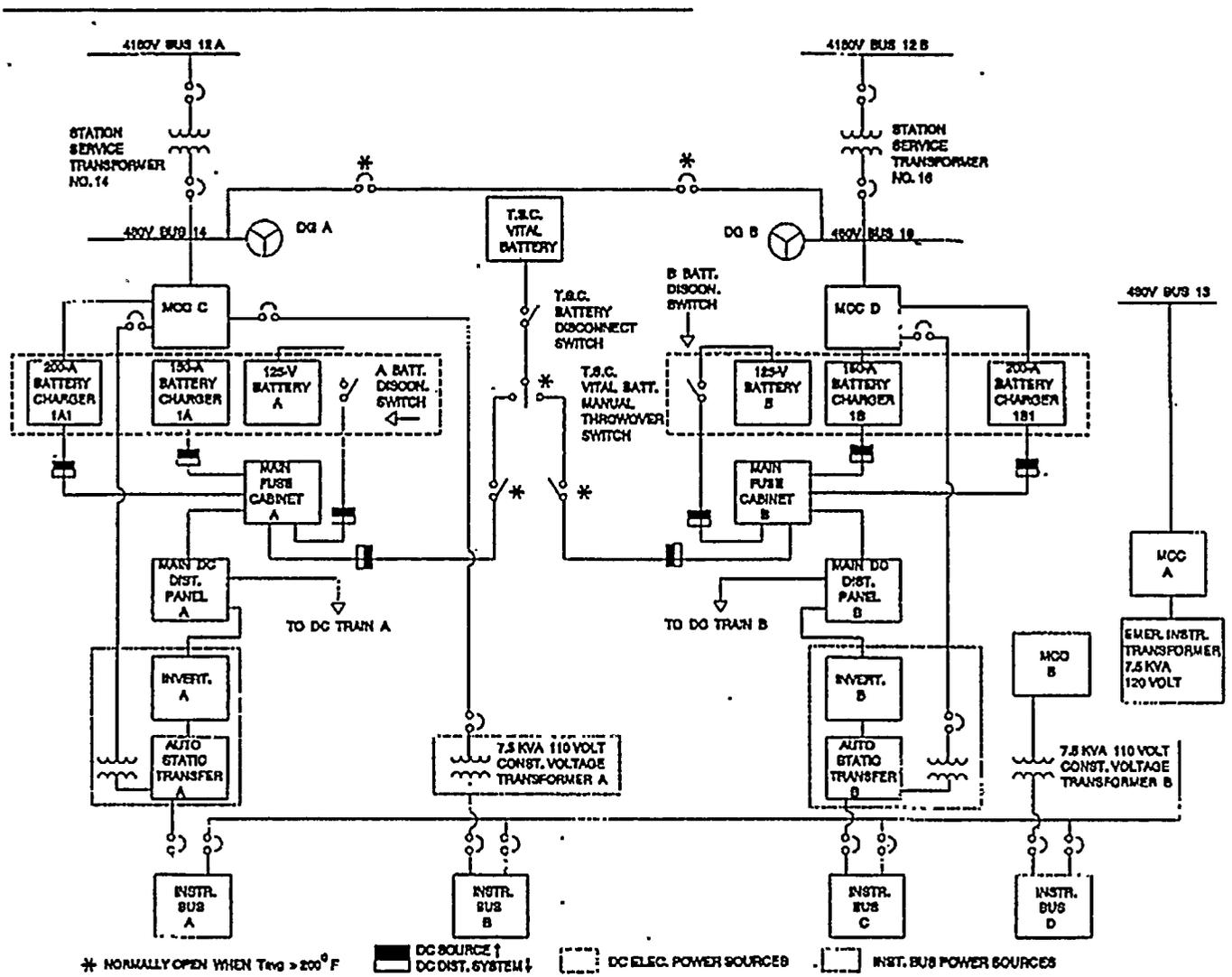
This SR is modified by a Note stating that this SR shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution system and challenge safety systems.

(continued)

BASES (continued)

- REFERENCES
1. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
 2. UFSAR, Section 8.3.2.
 3. UFSAR, Section 9.4.9.3.
 4. UFSAR, Chapter 6.
 5. UFSAR, Chapter 15.
 6. UFSAR, Section 8.3.1.
 7. 10 CFR 50, Appendix A, GDC 17.
 8. IEEE-450-1980.
 9. Regulatory Guide 1.32, February 1977.
 10. Regulatory Guide 1.129, December 1974.
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Figure B 3.8.4-1



B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.5, "DC Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6, the number of required DC electrical sources may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the DC electrical sources, must be removed from service. The minimum required DC electrical sources is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 ensures that:

- a. Required features needed to mitigate a fuel handling accident are available;
- b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the DC electrical power sources ensures that one train of DC sources are OPERABLE in the event of:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The DC electrical power sources are required to be OPERABLE to support the distribution subsystems required OPERABLE by LCO 3.8.10, "Distribution Systems - MODES 5 and 6." If only one DC electrical power distribution train is required to be OPERABLE, the minimum source consists of a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the required train. If both DC electrical power trains are required, one DC source must contain a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the train system. The second DC source may consist of only a battery charger with a capacity of at least 150 amps, or a battery, and the corresponding control equipment and interconnecting cabling. The two must be sufficiently independent that a loss of all offsite power sources, a loss of onsite standby power, or a worst case single failure does not affect more than one required DC electrical power train. This ensures the availability of sufficient DC electrical power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The AC powered and DC powered fan ventilation units are not required to be OPERABLE for this LCO, but some form of ventilation may be required to meet SR 3.8.6.4 and SR 3.8.6.5.

(continued)



BASES (continued)

APPLICABILITY The DC electrical power sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the affects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4."

ACTIONS

A.1

Although two trains may be required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," the remaining DC electrical train may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS, and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable associated with the required inoperable DC power source(s), appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. Required features remaining powered from a DC electrical source, even if that source is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

(continued)



BASES

ACTIONS
(continued)

A.2.1, A.2.2, A.2.3, and A.2.4

With one or more required DC electrical power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

It is further required to immediately initiate action to restore the required DC electrical power source and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE
REQUIREMENTS

SR 3.8.5.1

This SR requires the performance of SRs from LCO 3.8.4 that are necessary for ensuring the OPERABILITY of the DC electrical power subsystem in MODES 5 and 6.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.5.1 (continued)

This SR precludes requiring the OPERABLE DC electrical power source from being removed from service to perform a battery service test or a performance discharge test. With limited DC sources available, a single event could compromise multiple required safety features. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DC electrical power source is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.4 for a discussion of the specified SR.

REFERENCES

None.



B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

Each DC electrical power train contains a 125 VDC battery which is capable of carrying the expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 105 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 1). The batteries are normally in standby since the associated battery chargers provide for the required DC system loads.

The batteries for Train A and Train B DC electrical power are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and 100% design demand. Battery size is based on 125% of required capacity for aging considerations.

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries to ensure that the batteries are capable of performing their safety function as required by LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4," and LCO 3.8.5, "DC Sources - MODES 5 and 6."

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses assume Engineered Safety Feature systems are OPERABLE (Refs. 2 and 3). The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation. The DC sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

Battery cell parameters satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that battery cell parameters for Train A and B batteries be within limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery cell parameters are defined for electrolyte level, temperature, float voltage, and specific gravity. The limits for electrolyte level, float voltage, and specific gravity are conservatively established for both designated pilot cells and connected cells within plant procedures. Failure to meet these established limits may allow continued DC electrical system function provided that the limit specified in the associated Surveillance Requirement for each connected cell is not exceeded. The term "connected cell" excludes any battery cell that may be jumpered out.

(continued)

BASES (continued)

APPLICABILITY The battery cell parameters for Train A and Train B batteries are required solely for the support of the associated DC electrical power subsystem. Therefore, the battery cell parameter limits are required to be met when the DC power source is required to be OPERABLE. Since the Train A and Train B batteries support LCO 3.8.4 and LCO 3.8.5, the battery cell parameters are required to be met in MODES 1, 2, 3, and 4, and when the associated DC electrical power subsystems are required to be OPERABLE in MODES 5 and 6.

ACTIONS The ACTIONS are modified by a Note to provide clarification that separate condition entry is allowed for each battery. Separate Condition entry is acceptable since the battery cell parameters are provided on a battery basis.

A.1

With one or more batteries with one or more battery cell parameters outside the limits for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power train must be immediately declared inoperable and actions taken per LCO 3.8.4 or LCO 3.8.5.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.6.1

This SR verifies that the electrolyte level of each connected battery cell is above the top of the plates and not overflowing. This is consistent with IEEE-450 (Ref. 4) and ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Frequency of 31 days is consistent with IEEE-450.

SR 3.8.6.2

This SR verifies that the float voltage of each connected battery cell is > 2.07 V. This limit is based on IEEE-450 (Ref. 4) which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement. The frequency of 31 days is also consistent with IEEE-450.

SR 3.8.6.3

This SR verifies the specific gravity of the designated pilot cell in each battery is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. These values are based on manufacturer recommendations. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is further discussed in IEEE-450. The Frequency of 31 days is consistent with IEEE-450.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.6.4

This SR verifies the average electrolyte temperature of the designated pilot cell in each battery is $\geq 55^{\circ}\text{F}$. This temperature limit is an initial assumption of the battery capacity calculations. The Frequency of 31 days is consistent with IEEE-450 (Ref. 4).

SR 3.8.6.5

This SR verifies that the average temperature of every fifth cell of each battery is $\geq 55^{\circ}\text{F}$. This is consistent with the recommendations of IEEE-450 (Ref. 4). Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. The Frequency of 92 days is consistent with IEEE-450.

SR 3.8.6.6

This SR verifies the specific gravity of each connected cell is not more than 0.020 below average of all connected cells and that the average of all connected cells is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. These values are based on manufacturer recommendations and IEEE-450 (Ref. 4) which ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that discussed for SR 3.8.6.3. The Frequency of 92 days is consistent with IEEE-450.

REFERENCES

1. UFSAR, Section 3.8.2.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. IEEE-450-1980.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 AC Instrument Bus Sources—MODES 1, 2, 3, and 4

BASES

BACKGROUND

The AC instrument bus electrical power distribution subsystem consists of four 120 VAC instrument buses. The power source for one 120 VAC instrument bus (Instrument Bus D) is normally supplied from offsite power via a non-Class 1E constant voltage transformer (CVT) such that only three buses are considered safety related (see Figure 3.8.4-1). These three 120 VAC instrument buses (A, B, and C) supply a source of power to instrumentation and controls which are used to monitor and actuate the Reactor Protection System (RPS) and Engineered Safety Features (ESF) and other components (Ref. 1). The loss of Instrument Bus D is addressed in LCO 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and LCO 3.3.3, "Post-Accident Monitoring Instrumentation."

Instrument Buses A and C can be supplied power either from inverters which are powered from separate and redundant DC power sources, a non-Class 1E CVT (maintenance CVT) powered from offsite power, or a Class 1E CVT (see Figure B 3.8.4-1). The inverters are the preferred source of power for Instrument Bus A and C because of the stability and reliability they achieve.

Instrument Bus B can be supplied power from either a Class 1E CVT or a non-Class 1E CVT (maintenance CVT) powered from offsite power. The Class 1E CVT, supplied by motor control center C (MCC C is supplied by 480 V safeguards Bus 14), is the preferred source of power for Instrument Bus B because of the potential to have a power interruption if offsite power were unavailable.

(continued)

BASES

BACKGROUND
(continued)

The majority of instrumentation and controls supplied by the 120 VAC instrument buses are fail safe devices such that they go to their post accident position upon loss of power. However, a notable exception to this is the actuation logic for Containment Spray (CS) System which requires 120 VAC and 125 VDC power in order to function. This prevents a spurious CS actuation from occurring if control power were lost. The actuation logic for CS is powered from all three instrument buses and from both DC electrical power distribution trains.

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 2 and 3), assume Engineered Safety Feature systems are OPERABLE. The AC instrument bus power sources are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESF instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC instrument bus power sources is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the plant. This includes maintaining required AC instrument buses OPERABLE in the event of:

- a. An assumed loss of all offsite AC electrical power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC instrument bus power sources ensures that one train of AC instrument buses are available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one AC instrument bus power source).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of AC instrument bus power also ensures that at least one train of AC instrument buses is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 4). Therefore, the requirements of GDC 17 (Ref. 5) can be met at all times.

The AC instrument bus sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The AC instrument bus sources ensure the availability of 120 VAC electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESF instrumentation and controls is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source for the third instrument bus.

(continued)

BASES

LCO
(continued)

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4").

For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4").

APPLICABILITY

The AC instrument bus power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOs or abnormal transients; and
- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

AC instrument bus power requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "AC Instrument Bus Sources - MODES 5 and 6."

ACTIONS

A.1, A.2, and A.3

With an inverter inoperable, its associated AC instrument bus becomes inoperable until it is re-energized from either its Class 1E or non-Class 1E CVT.

Required Action A.1 allows the instrument bus to be powered from either its associated Class 1E CVT or from a non-Class 1E CVT. For Instrument Buses A and C, the non-Class 1E power is supplied by a non-safety related motor control center (MCC A) which is supplied by 480 V Bus 13. The Completion Time of 2 hours is consistent with LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4".

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Required Action A.2 is intended to limit the amount of time that the instrument bus can be connected to a non-Class 1E power supply. The 24 hour Completion Time is based upon engineering judgement, taking into consideration the time required to repair the Class 1E CVT or the inverter and the additional risk to which the plant is exposed because of the connection to a non-Class 1E power supply.

Required Action A.3 allows 72 hours to fix the inoperable inverter and restore it to OPERABLE status. The 72 hour Completion Time is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC instrument bus is powered from its CVT, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible, battery backed inverter source to the AC instrument buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

With the Class 1E CVT for Instrument Bus B inoperable, the instrument bus becomes inoperable until it is re-energized from its non-Class 1E CVT. Required Action B.1 requires Instrument Bus B to be powered from its non-Class 1E CVT within 2 hours. The non-Class 1E power is supplied by a nonsafety related 480 V motor control center (MCC A) which is supplied by 480 V Bus 13.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Required Action B.2 is intended to limit the amount of time that Instrument Bus B can be connected to a non-Class 1E power supply. The 7 day limit is based on engineering judgement, taking into consideration the time required to repair the Class 1E CVT and the additional risk to which the plant is exposed because of the Class 1E CVT inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When Instrument Bus B is powered from its non-Class 1E CVT, it is relying upon interruptible offsite AC electrical power sources. The Class 1E, diesel generator backed, CVT to Instrument Bus B is the preferred power source for powering instrumentation trip setpoint devices.

C.1 and C.2

If the inoperable devices or components cannot be restored to OPERABLE status or other Required Actions are not completed within the required Completion Time of Condition A or B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If two or more required AC instrument bus power sources are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when both inverters, or one or more inverters and the Class 1E CVT to Instrument Bus B are discovered to be inoperable.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.7.1

This SR verifies correct static switch alignment to Instrument Bus A and C. This verifies that the inverters are functioning properly and AC Instrument Bus A and C are energized from their respective inverter. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument buses. The Frequency of 7 days takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

SR 3.8.7.2

This SR verifies the correct Class 1E CVT alignment to Instrument Bus B. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

1. UFSAR, Chapter 8.3.2.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 8.3.1.
 5. 10 CFR 50, Appendix A, GDC 17.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 AC Instrument Bus Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.7, "AC Instrument Bus Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6, the number of required AC instrument buses may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the AC instrument bus sources, must be removed from service. The minimum required AC instrument bus electrical subsystem is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum AC instrument bus power sources to each required AC instrument bus during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC instrument bus power sources ensures that one train of the AC instrument buses are OPERABLE in the event of:

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC instrument bus power sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESF instrumentation and controls is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source for the third instrument bus.

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4).

(continued)



BASES

LCO
(continued) For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4). Power sources ensure the availability of sufficient power to the required AC instrument buses to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY The inverters required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available.

AC Instrument Bus power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1

Although two trains may be required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," the remaining OPERABLE AC instrument bus train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable with the associated AC instrument bus power source inoperable, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. This condition must be entered when the inverters for Instrument Bus A or C are inoperable, or the Class 1E CVT for Instrument Bus B is inoperable.

(continued)



BASES

ACTIONS
(continued)

A.2.1, A.2.2, A.2.3, and A.2.4

With one or more required AC instrument bus power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

It is further required to immediately initiate action to restore the required AC instrument bus power source and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC instrument bus power source should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from an alternate power source.

SURVEILLANCE
REQUIREMENTS

SR 3.8.8.1

This SR verifies correct static switch alignment to the required AC instrument buses. This SR verifies that the inverter is functioning properly and the AC instrument bus is energized from the inverter. The verification ensures that the required power is available for the instrumentation connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant capability of the inverter and other indications available in the control room that alert the operator to inverter malfunctions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.8.2

This SR verifies the correct Class 1E CVT alignment when Instrument Bus B is required. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. 3 The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential action components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for each of these electrical power sources which are further divided and organized based on voltage considerations and safety classification. This LCO is provided to specify the AC, DC, and AC instrument bus power electrical power distribution subsystems which are required to supply safety related and Engineered Safety Feature (ESF) Systems in MODES 1, 2, 3, and 4.

The onsite Class 1E AC, DC, and AC instrument bus electrical power distribution subsystems are each divided into two redundant and independent distribution trains. Each of these electrical power distribution subsystems, and their trains, are discussed in detail below.

(continued)

BASES

BACKGROUND
(continued)

AC Electrical Power Distribution Subsystem

The Class 1E AC electrical power distribution subsystem is organized into two redundant and independent trains (Train A and Train B). Each train consists of two 480 V safeguards buses, distribution panels, motor control centers and load centers (see Figure B 3.8.1-1). The 480 V safeguards buses for each train are capable of being supplied from two sources of offsite power as well as a dedicated onsite emergency diesel generator (DG) source. These power sources are discussed in more detail in the Bases for LCO 3.8.1, "AC Sources—MODES 1, 2, 3, and 4." The 480 V safeguards buses in turn supply motor control centers, distribution panels and load centers which supply motive power to required motor operated valves, pumps, dampers, or any other component which requires AC power to perform its safety related function. The AC electrical power distribution subsystem also supplies one of the three required AC instrument buses through a constant voltage transformer and provides a backup source for the other two instrument buses. The list of all required AC 480 V safeguards buses is provided in Table B 3.8.9-1.

DC Electrical Power Distribution Subsystem

The Class 1E DC electrical power distribution subsystem is organized into two redundant and independent trains (Train A and Train B). Each train consists of a Class 1E battery and two battery chargers (with a charging capacity of at least 150 amps) which supply a main 125 VDC distribution panel (see Figure B 3.8.4-1). These power sources are discussed in more detail in the Bases for LCO 3.8.4, "DC Sources—MODES 1, 2, 3, and 4." Each main distribution panel supplies secondary distribution panels which provide control power to AC powered components and control power for other devices such as solenoid operated valves and air operated valves. The DC electrical power distribution subsystem also supplies two of the four AC instrument buses through inverters. The list of all required DC distribution panels is provided in Table B 3.8.9-1.

(continued)

BASES

BACKGROUND
(continued)

AC Instrument Bus Electrical Power Distribution Subsystem

The AC instrument bus electrical power distribution subsystem consists of four 120 VAC instrument buses. The power source for one 120 VAC instrument bus (Instrument Bus D) is supplied from offsite power via a non Class 1E constant voltage transformer (CVT) such that only three buses are considered safety related (see Figure B 3.8.4-1). These three buses are organized into two redundant and independent trains (Train A and Train B). These trains supply a source of power to instrumentation and controls which are used to monitor and actuate ESF and other components. Train A consists of two buses with one bus (Instrument Bus A) normally powered from an inverter and the other (Instrument Bus B) normally powered from a Class 1E CVT. Train B consists of one bus (Instrument Bus C) normally powered from an inverter. The long-term alternate power supplies for Instrument Bus A and C are two Class 1E CVTs, each powered from the same train as the associated battery chargers, and their use is governed by LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4." The list of required 120 VAC instrument buses is provided in Table B 3.8.9-1. The loss of Instrument Bus D is addressed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.3, "Post-Accident Monitoring (PAM) Instrumentation."

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 1 and 2) assume ESF systems are OPERABLE. The AC, DC, and AC instrument bus electrical power distribution subsystems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining power distribution subsystems OPERABLE in the event of:

- a. An assumed loss of all AC offsite power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC, DC, and AC instrument bus electrical power distribution subsystems ensures that one train of each distribution subsystem is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one train of offsite standby AC power).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of offsite power also ensures that at least one AC, DC, and AC instrument bus train is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater train during the estimated 8 hours required to provide this power transfer (Ref. 3). Therefore, the requirements of GDC 17 (Ref. 4) can be met at all times.

The AC, DC, and AC instrument bus electrical power distribution subsystems satisfy Criterion 3 of the NRC Policy Statement.

LCO

Train A and Train B of the AC, DC, and AC instrument bus electrical power distribution subsystems are required to be OPERABLE. The power distribution subsystems and their trains listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

(continued)

BASES

LCO
(continued)

OPERABLE AC, DC, and AC instrument bus electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. Maintaining the Train A and Train B AC, DC, and AC instrument bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not compromised. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

Tie breakers between redundant safety related AC, DC, and AC instrument bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s).

If any of the following listed tie breakers are closed, the affected redundant electrical power distribution subsystem is considered inoperable. This does not, however, preclude AC buses from being powered from the same offsite circuit.

a. AC power 480 V safeguards bus tie breakers (Ref. 5)

- Bus-Tie 14-16
- Bus-Tie 16-14
- Bus-Tie 17-18
- Bus-Tie 16-15
- Bus-Tie 14-13

b. DC control power automatic throwover switches (in normal position) (Ref. 6)

- DG Control Panel A
- DG Control Panel B
- Bus 14 Control Power and Undervoltage Cabinet
- Bus 16 Control Power and Undervoltage Cabinet
- Bus 17 Control Power and Undervoltage Cabinet
- Bus 18 Control Power and Undervoltage Cabinet

(continued)



BASES

LCO
(continued)

- c. Technical Support Center battery connections to DC power Battery A and B (Ref. 6)

TSC/Battery A Fused Disconnect Switch
TSC/Battery B Fused Disconnect Switch

The trains as specified in Table B 3.8.9-1 only identify the major AC, DC, and AC instrument bus electrical power distribution subsystem components. A train is defined to begin from the boundary of the power source for the respective subsystem (as defined in the power source LCOs), and continues up to the isolation device for the supplied safety related or ESF component (e.g., safety injection pump). The isolation device for the supplied safety related or ESF component is only considered part of the train when the device is not capable of opening to isolate the failed component from the train (e.g., breaker unable to open an overcurrent). Otherwise, the failure of the isolation device to close to provide power to the component is addressed by the respective component's LCO. The isolation device for nonsafety related components are considered part of the train since these devices must be available to protect the safety related functions. Therefore, the train boundary essentially ends at the motor control center or bus which supplies multiple components.

The inoperability of any component within the above defined train boundaries renders the train inoperable.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

(continued)

BASES

APPLICABILITY
(continued)

Electrical power distribution subsystem requirements for MODES 5 and 6 are addressed in LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

ACTIONS

A.1

With one AC electrical power distribution train inoperable, the remaining AC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining AC power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels which comprise a train must be restored to OPERABLE status within 8 hours.

The worst case Condition A scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the plant is more vulnerable to a complete loss of AC power.

The Completion Time for restoring the inoperable train before requiring a plant shutdown is limited to 8 hours because of:

- a. The potential for decreased safety if the plant operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE train with AC power which results in the loss of multiple safety functions.

(continued)

BASES

ACTIONS
(continued)

B.1

With one AC instrument bus electrical power distribution train inoperable, the remaining OPERABLE AC instrument bus train is capable of supporting the minimum safety functions necessary to shut down the plant and maintain it in the safe shutdown condition. Overall reliability is reduced, however, because a single failure in the remaining AC instrument bus train could result in the minimum ESF functions not being supported. Therefore, the AC instrument bus train must be restored to OPERABLE status within 2 hours.

Condition B represents one AC instrument bus train without power which includes the potential loss of both the DC source and the associated AC sources to the instrument bus. In this situation, the plant is significantly more vulnerable to a complete loss of all noninterruptible power. Therefore, the Completion Time is limited to 2 hours due to the potential vulnerabilities. Taking exception to LCO 3.0.2 for components without adequate 120 VAC power, that would have Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

- a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate 120 VAC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and
- c. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE AC instrument bus train.

(continued)

BASES

ACTIONS
(continued)

B.1 (continued)

The 2 hour Completion Time takes into account the importance to safety of restoring the AC instrument bus train to OPERABLE status, the redundant capability afforded by the other OPERABLE instrument bus train, and the low probability of a DBA occurring during this period.

C.1

With one DC electrical power distribution train inoperable, the remaining DC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required DC distribution panels must be restored to OPERABLE status within 2 hours.

Condition C represents one train without adequate DC power (e.g., the battery and required battery charger are inoperable). In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. Therefore, the Completion Time is limited to 2 hours due to this potential vulnerability. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

(continued)



BASES

ACTIONS

C.1 (continued)

- c. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE train with DC power.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With two trains with inoperable electrical power distribution subsystems, the potential for a loss of safety function is greater. If a loss of safety function exists, no additional time is justified for continued operation and LCO 3.0.3 must be entered. This Condition may be entered with the loss of two trains of the same electrical power distribution subsystem, or with loss of Train A of one electrical power distribution subsystem coincident with the loss of Train B of a second electrical power distribution subsystem such that a loss of safety function exists.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.9.1

This SR verifies that the electrical power trains are functioning properly, with all required power source circuit breakers closed, tie-breakers open, and the buses energized from their allowable power sources. Required voltage for the AC electrical power distribution subsystem is ≥ 420 VAC; for the DC electrical power distribution subsystem, ≥ 108.6 VDC; and for AC instrument bus electrical power distribution subsystem, between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the redundant capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
 2. UFSAR, Chapter 15.
 3. UFSAR, Section 8.3.1.
 4. 10 CFR 50, Appendix A, GDC 17.
 5. UFSAR, Figure 8.3-1.
 6. UFSAR, Figure 8.3-6.
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Table B 3.8.9-1 (page 1 of 1)
 AC and DC Electrical Power Distribution Systems

DISTRIBUTION SUBSYSTEM	VOLTAGE	TRAIN A	TRAIN B
AC Power	480 V	Bus 14 Bus 18	Bus 16 Bus 17
DC Power	125 V	Main DC Fuse Cabinet A (DCPDPCB02A) Main DC Distribution Panel A (DCPDPCB03A) Aux Bldg DC Distribution Panel A (DCPDPA01A) Aux Bldg DC Distribution Panel A1 (DCPDPA02A) DG A DC Distribution Panel A (DCPDPA01A) Screenhouse DC Distribution Panel A (DCPDPSH01A) MCB DC Distribution Panel A (DCPDPCB04A)	Main DC Fuse Cabinet B (DCPDPCB02B) Main DC Distribution Panel B (DCPDPCB03B) Aux Bldg DC Distribution Panel B (DCPDPA01B) Aux Bldg DC Distribution Panel B1 (DCPDPA02B) DG B DC Distribution Panel B (DCPDPOG01B) Screenhouse DC Distribution Panel B (DCPDPSH01B) MCB DC Distribution Panel B (DCPDPCB04B) Turbine Bldg DC Distribution Panel (DCPDPTB01B)
AC Instrument Bus	120 V	Bus A Bus B	Bus C

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

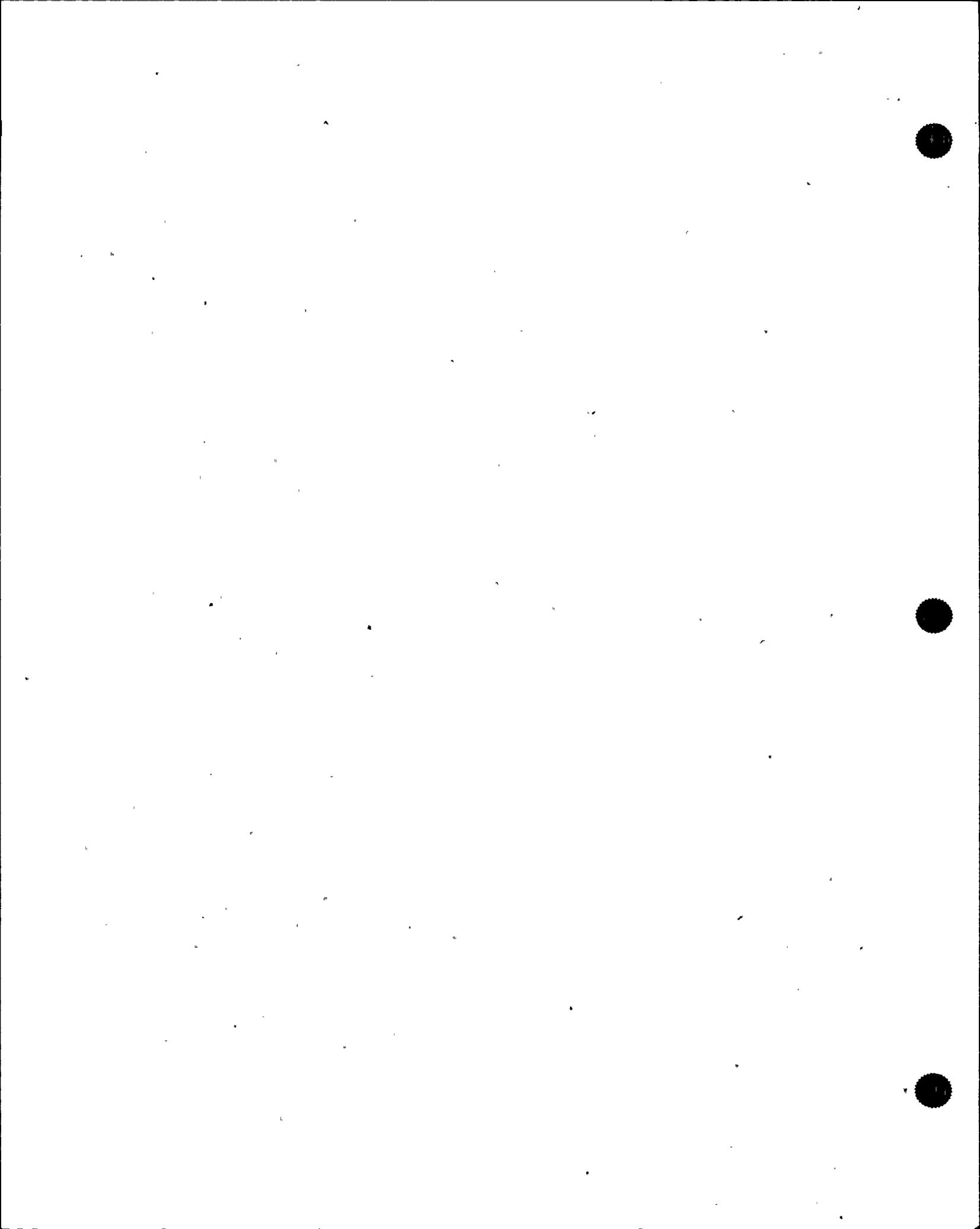
In MODES 5 or 6, the number of required AC, DC, and AC instrument bus electrical power distribution subsystems, or the number of required trains within these electrical power distribution subsystems may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the electrical power distribution subsystems, must be removed from service.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC, DC, and AC instrument bus electrical power distribution subsystems during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition and refueling condition.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution subsystems ensures that one train of the onsite power or offsite AC sources are OPERABLE in the event of:

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6 this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC, DC, and AC instrument bus electrical power distribution subsystems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO Various combinations of AC, DC, and AC instrument bus electrical power distribution subsystems, trains within these subsystems, and equipment and components within these trains are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

The LCOs which apply when the Reactor Coolant System is $\leq 200^{\circ}\text{F}$ and which may require a source of electrical power are:

- LCO 3.1.1 SHUTDOWN MARGIN (SDM)
- LCO 3.3.1 Reactor Trip System (RTS) Instrumentation
- LCO 3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation
- LCO 3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation
- LCO 3.4.7 RCS Loops - MODE 5, Loops Filled
- LCO 3.4.8 RCS Loops - MODE 5, Loops Not Filled
- LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System
- LCO 3.7.9 Control Room Emergency Air Treatment System (CREATS)
- LCO 3.9.2 Nuclear Instrumentation
- LCO 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft
- LCO 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

Maintaining the necessary trains of the AC, DC, and AC instrument bus electrical power distribution subsystems energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

(continued)



BASES (continued)

APPLICABILITY

The AC, DC, and AC instrument bus electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a postulated event and maintain the plant in the cold shutdown or refueling condition are available.

The AC, DC, and AC instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4."

ACTIONS

A.1

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations involving positive reactivity additions. By allowing the option to declare required features associated with an inoperable distribution subsystem or train inoperable, appropriate restrictions are implemented in accordance with the LCO ACTIONS of the affected required features.

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

With one or more required electrical power distribution subsystems or trains inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position of normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

It is further required to immediately initiate action to restore the required AC, DC, and AC instrument bus electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

In addition to performance of the above conservative Required Actions, a required residual heat removal (RHR) loop may be inoperable. In this case, Required Actions A.2.1, A.2.2, A.2.3, and A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 requires declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the electrical power distribution trains are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

None.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LC0 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend positive reactivity additions.	Immediately
B. Two source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u> B.2 Perform SR 3.9.1.1.	4 hours <u>AND</u> Once per 12 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. No audible count rate.	C.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	C.3 Perform SR 3.9.1.1	4 hours
		<u>AND</u>
		Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.2.1 Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2 -----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	24 months



3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

- a. The equipment hatch shall be either:
 - 1. bolted in place with at least one access door closed, or
 - 2. isolated by a closure plate that restricts air flow from containment;
- b. One door in the personnel air lock shall be closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

-----NOTE-----
 The required RHR loop may be removed from operation for \leq 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System (RCS) boron concentration.

APPLICABILITY: MODE 6 with the water level \geq 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in RCS boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.1 Verify one RHR loop is in operation and circulating reactor coolant.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—Water Level < 23 Ft

LCO 3.9.5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore RHR loop(s) to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR loop in operation.	B.1 Suspend operations involving a reduction in Reactor Coolant System boron concentration.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>	
	B.3 Close all containment penetrations providing direct access from containment to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.1 Verify one RHR loop is in operation and circulating reactor coolant.	12 hours
SR 3.9.5.2 Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

3.9 REFUELING OPERATIONS

3.9.6 Refueling Cavity Water Level

LC0 3.9.6 Refueling cavity water level shall be maintained \geq 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment,
During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.6.1 Verify refueling cavity water level is \geq 23 ft above the top of reactor vessel flange.	24 hours



B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration ensures the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the filled portions of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity that are hydraulically coupled to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant. The refueling boron concentration limit is specified in the Core Operation Limits Report (COLR). Plant refueling procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{\text{eff}} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant refueling procedures.

Atomic Industrial Forum (AIF) GDC 27 requires that two independent reactivity control systems preferably of different design principles be provided (Ref. 1). In addition to the reactivity control achieved by the control rods, reactivity control is provided by the chemical and volume control system (CVCS) which regulates the concentration of boric acid solution (neutron absorber) in the RCS. The CVCS is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes which may stress or damage the fuel beyond allowable limits.

The reactor is brought to shutdown conditions (i.e., MODE 5) before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized the vessel head is unbolted and removed. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by use of the Residual Heat Removal (RHR) System pumps.

(continued)



BASES

BACKGROUND
(continued)

The pumping action of the RHR System into the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity provide mixing for the borated coolant in the refueling canal.

The RHR System is in operation during refueling (see LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level \geq 23 Ft," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level $<$ 23 Ft") to provide forced circulation in the RCS and assist in maintaining the boron concentration in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE
SAFETY ANALYSES

During refueling operations, two types of accidents can occur within containment that affect the fuel and require control of reactivity. These two accident types are a fuel handling accident and a boron dilution event. Both accidents assume that initial core reactivity is at its highest (i.e., at the beginning of the fuel cycle or the end of refueling).

A fuel handling accident can occur during fuel movement in the reactor vessel, the refueling canal, or the refueling cavity and includes a dropped fuel assembly and an incorrectly transferred fuel assembly. The most limiting fuel handling accident is a dropped fuel assembly which is dropped adjacent to other fuel assemblies such that it results in the largest exposure of fuel in the dropped assembly. The negative reactivity effect of the soluble boron compensates for the increased reactivity for both types of accidents. Hence, the boron concentration ensures that $k_{\text{eff}} \leq 0.95$ (i.e., 5% $\Delta k/k$ SHUTDOWN MARGIN) during the refueling operation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The second type of accident is a boron dilution event which results from inadvertent addition of unborated water to the RCS, refueling cavity, and refueling canal. The assumptions used in the boron dilution event (Ref. 2) provide for a maximum dilution flow of 120 gpm through two charging pumps (i.e., 60 gpm per pump) using unborated water as supplied by the two reactor makeup water pumps (60 gpm per pump). The RCS is also assumed to be at low water levels, uniformly mixed by the RHR System, with the minimum boron concentration as specified in the COLR. The operator has prompt and definite indication of significant boron dilution from an audible count rate function provided by the source range neutron flux instrumentation (see LCO 3.9.2, "Nuclear Instrumentation"). The increased count rate is a function of the effective subcritical multiplication factor. The results of this analysis conclude that an operator has at least 48.8 minutes before SHUTDOWN MARGIN is lost and the reactor goes critical which is sufficient time for operators to mitigate this event. This time is also greater than the 30 minutes required by Reference 3 for dilution events during refueling. Isolating the boron dilution source is performed by closing valves and/or stopping the reactor makeup water pumps.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that a minimum boron concentration be maintained in the refueling canal, the refueling cavity and the portions of the RCS that are hydraulically coupled with the reactor core while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations and that a core k_{eff} of < 1.0 is maintained during a boron dilution event. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

(continued)

BASES (continued)

APPLICABILITY This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{\text{eff}} \leq 0.95$ during fuel handling operations. In MODES 1 and 2 with $k_{\text{eff}} \geq 1.0$, LCO 3.1.4, "Rod Group Alignment Limit," LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits" ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 2 with $k_{\text{eff}} < 1.0$ and MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures an adequate amount of negative reactivity is available to maintain the reactor subcritical.

ACTIONS A.1, A.2, and A.3

If the boron concentration of the filled portions of the RCS, the refueling canal, and the refueling cavity hydraulically coupled to the reactor core, is less than its limit, an inadvertent criticality may occur due to a boron dilution event or incorrect fuel loading. To minimize the potential of an inadvertent criticality resulting from a fuel loading error or an operation that could cause a reduction in boron concentration, CORE ALTERATIONS and positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions (i.e., other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures) shall not preclude moving a component to a safe position.

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Once action has been initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE
REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration of the refueling canal, the refueling cavity, and the portions of the RCS that are hydraulically coupled, is within the COLR limits. The boron concentration of the coolant is determined by chemical analysis. The sample should be representative of the portions of the RCS, the refueling canal, and the refueling cavity that are hydraulically coupled with the reactor core.

A Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, Issued for comment July 10, 1967.
 2. UFSAR, Section 15.4.4.2.
 3. NUREG-0800, Section 15.4.6.
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B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors (N-31 and N-32) are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux detectors are proportional counters that are filled with boron trifluoride (BF_3) gas (Ref. 1). The detectors monitor the neutron flux in counts per second and provide continuous visual indication in the control room. Audible count rate is also available in the control room from either of the source range neutron flux monitors to alert operators to a possible boron dilution event. The NIS is designed in accordance with the criteria presented in Reference 2.

APPLICABLE
SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide redundant indication to alert operators of unexpected changes in core reactivity. An increase in the audible count rate alerts the operators that a boron dilution event is in progress. Sufficient time is available for the operator to recognize the increase in audible count rate and to terminate the event prior to a loss of SHUTDOWN MARGIN (see Bases for LCO 3.9.1, "Boron Concentration"). Isolating the boron dilution source is performed by closing valves and stopping reactor makeup water pumps.

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO This LCO requires two source range neutron flux monitors be OPERABLE to ensure redundant monitoring capability is available to detect changes in core reactivity.

To be OPERABLE, each monitor must provide visual indication and at least one of the two monitors must provide an audible count rate function in the control room.

APPLICABILITY In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity conditions in this MODE. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions A.1 and A.2 shall not preclude completion of movement of a component to a safe position (i.e., other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures).

B.1 and B.2

With no source range neutron flux monitor OPERABLE there are no direct means of detecting changes in core reactivity. Therefore, actions to restore a monitor to OPERABLE status shall be initiated immediately and continue until a source range neutron flux monitor is restored to OPERABLE status.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Since CORE ALTERATIONS and positive reactivity additions are not to be made per Required Actions A.1 and A.2, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

C.1, C.2, and C.3

With no audible count rate available, only visual indication is available and prompt and definite indication of a boron dilution event has been lost. Therefore, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position or that is a normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion time of 4 hours is sufficient to obtain and analyze coolant samples for boron concentration. The Frequency of once per 12 hours ensures unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.2.1

This SR is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one monitor to a similar parameter on another monitor. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range monitors, but each monitor should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

SR 3.9.2.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 7.7.3.2.
 2. Atomic Industrial Forum (AIF) GDC 13 and 19, Issued for Comment July 10, 1967.
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B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 5, there are no accidents of concern which require containment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be bolted in place. Good engineering practice dictates that a minimum of 4 bolts be used to hold the equipment hatch in place and that the bolts be approximately equally spaced. As an alternative, the equipment hatch can be isolated by a closure plate that restricts air flow from containment.

(continued)

BASES

BACKGROUND
(continued)

The containment equipment and personnel air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of plant shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed in the personnel and equipment hatch (unless the equipment hatch is isolated by a closure plate).

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The Shutdown Purge System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a Mini-Purge System, includes a 6 inch purge penetration and a 6 inch exhaust penetration. During MODES 1, 2, 3, and 4, the shutdown purge and exhaust penetrations are isolated by a blind flange with two O-rings that provide the necessary boundary. The two air operated valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation Instrumentation System. Neither of the subsystems is subject to a Specification in MODE 5.

(continued)

BASES

BACKGROUND
(continued)

In MODE 6, large air exchangers are used to support refueling operations. The normal 36 inch Shutdown Purge System is used for this purpose, and each air operated valve is closed by the Containment Ventilation Isolation Instrumentation in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation."

The Mini-Purge System also remains operational in MODE 6, and all four valves are also closed by the Containment Ventilation Isolation Instrumentation.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE
SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed using the criteria of Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in 10 CFR 100. Standard Review Plan (SRP), Section 15.7.4, Rev. 1 (Ref. 2), requires containment closure even though this is not an assumption of the accident analyses. The acceptance limits for offsite radiation exposure is 96 rem (Ref. 3).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement since these are assumed in the SRP.

(continued)



BASES (continued)

LCO This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that at least one valve in each of these penetrations is isolable by the Containment Ventilation Isolation System.

APPLICABILITY The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions, no requirements are placed on containment penetration status.

ACTIONS A.1 and A.2

If the containment equipment hatch (or its closure plate), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.7.
 2. NUREG-800, Section 15.7.4, Rev. 1, July 1981.
 3. Letter from D. M. Crutchfield, NRC, to J. Maier, RG&E,
Subject: "Fuel Handling Accident Inside Containment,"
dated October 7, 1981.
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B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level
 \geq 23 Ft

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), and to provide mixing of the borated coolant to prevent thermal and boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s) where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS loop "B" cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSES

The safety analysis for the boron dilution event during refueling assumes one RHR loop is in operation (Ref. 2). This initial assumption ensures continuous mixing of the borated coolant in the reactor vessel. The analysis also assumes the RCS is at equilibrium boron concentration and dilution occurs uniformly throughout the system. Therefore, thermal or boron stratification is not postulated. In order to ensure adequate mixing of the borated coolant, one loop of the RHR System is required to be OPERABLE, and in operation while in MODE 6, with water level \geq 23 ft above the top of the reactor vessel flange.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. Due to the water volume available in the RCS with a water level \geq 23 ft above the top of the reactor vessel flange, a significant amount of time exists before boiling of the coolant would occur following a loss of the required RHR pump. Since the loss of the required RHR pump results in the requirement to suspend operations involving a reduction in reactor coolant boron concentration, a boron dilution event is very unlikely. Therefore, this requirement dictates that single failures are not considered for this LCO due to the time available to operators to respond to a loss of the operating RHR pump.

The LCO permits de-energizing the required RHR pump for short durations provided no operations are permitted that would cause a reduction in the RCS boron concentration. This conditional de-energizing of the required RHR pump does not result in a challenge to the fission product barrier or result in coolant stratification.

RHR and Coolant Circulation-Water Level \geq 23 Ft satisfies criterion 2 of the NRC Policy Statement.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. One RHR loop is required to be OPERABLE and in operation to provide mixing of borated coolant to minimize the possibility of criticality.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg.

(continued)

BASES

LCO
(continued)

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allows the operator to view the core and permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles. This also permits operations such as RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity. Should both RHR loops become inoperable at anytime during operation in accordance with this Note, the Required Actions of this LCO should be immediately taken.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.5, "Refueling Cavity Water Level."

Requirements for the RHR System in MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.4, "RCS Loops-MODE 1 > 8.5% RTP;" LCO 3.4.5, "RCS Loops-MODES 1 \leq 8.5% RTP, 2 and 3;" LCO 3.4.6, "RCS Loops-MODE 4;" LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled;" and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." The RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Water Level < 23 Ft."

(continued)



BASES (continued)

ACTIONS

A.1, A.2, and A.3

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that could result in a reduction in the coolant boron concentration must be suspended immediately.

With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition. Therefore, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core.

With the plant in MODE 6 and the refueling water level \geq 23 ft above the top of the reactor vessel flange, removal of decay heat is by ambient losses only. Therefore, corrective actions shall be initiated immediately and shall continue until RHR loop requirements are satisfied..

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

(continued)

D
BASES (continued)

**SURVEILLANCE
REQUIREMENTS**

SR 3.9.4.1

This SR requires verification every 12 hours that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

REFERENCES

1. UFSAR, Section 5.4.5.
 2. UFSAR, Section 15.4.4.2.
-

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—Water Level
< 23 Ft

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), and to provide mixing of the borated coolant to prevent thermal and boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s) where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS loop "B" cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSES

The safety analysis for the boron dilution event during refueling assumes one RHR loop is in operation (Ref. 2). This initial assumption ensures continuous mixing of the borated coolant in the reactor vessel. The analysis also assumes the RCS is at equilibrium boron concentration and dilution occurs uniformly throughout the system. Therefore, thermal or boron stratification is not postulated.

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. This could lead to a loss of coolant in the reactor vessel. In addition, boiling of the coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of coolant and the reduction of boron concentration in the reactor coolant could eventually challenge the integrity of the fuel cladding, which is a fission product barrier.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In order to prevent a challenge to fuel cladding and to ensure adequate mixing of the borated coolant, two loops of the RHR System are required to be OPERABLE, and one loop in operation while in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange.

RHR and Coolant Circulation-Water Level < 23 Ft satisfies criterion 4 of the NRC Policy Statement.

LCO

Both RHR loops must be OPERABLE in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange. In addition, one RHR loop must be in operation in order to remove decay heat and provide mixing of borated coolant to minimize the possibility of criticality.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant.

Requirements for the RHR System in MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.4, "RCS Loops-MODE 1 > 8.5% RTP;" LCO 3.4.5, "RCS Loops-MODES 1 ≤ 8.5% RTP, 2 and 3;" LCO 3.4.6, "RCS Loops-MODE 4;" LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled;" and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." The RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-Water Level ≥ 23 Ft."

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.4, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1 and B.2

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. The potential for reduced boron concentrations by the addition of water with a lower boron concentration than that contained in the RCS must be reduced to prevent a criticality event. Therefore, operations involving a reduction in RCS boron concentration must be suspended immediately. Actions shall also be initiated immediately, and continued, to restore one RHR loop to operation. Since the plant is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

(continued)



BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.5.1

This SR requires verification every 12 hours that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.9.5.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 5.4.5.
 2. UFSAR, Section 15.4.4.2.
-



B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, requires a minimum water level of 23 ft above the top of the reactor vessel flange. This requirement ensures a sufficient level of water is maintained in the refueling cavity or portions hydraulically connected (e.g., refueling canal) to retain iodine fission product activity resulting from a fuel handling accident in containment (Ref. 1). The retention of iodine activity by the water limits the offsite dose from the accident well within the values specified in 10 CFR 100 (Ref. 2).

APPLICABLE
SAFETY ANALYSES

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 100 to be used in the accident analysis for iodine (Ref. 3). This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 3).

With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 2).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits and preserves the assumptions of the fuel handling accident analysis (Ref. 1). As such, it is the minimum required level during movement of fuel assemblies within containment. Maintaining this minimum water level in the refueling cavity also ensures that ≥ 23 ft of water is available in the spent fuel pool during fuel movement assuming that containment and Auxiliary Building atmospheric pressures are equal.

APPLICABILITY This LCO is applicable when moving irradiated fuel assemblies within containment. This LCO is also applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts. The LCO ensures a sufficient level of water is present in the refueling cavity to minimize the radiological consequences of a fuel handling accident in containment. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.11, "Spent Fuel Pool (SFP) Water Level."

ACTIONS A.1 and A.2

When the initial condition assumed in the fuel handling accident cannot be met, steps should be taken to preclude the accident from occurring. With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling cavity water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 15.7.3.3.
 2. 10 CFR 100.
 3. Regulatory Guide 1.25.
-

4.0 DESIGN FEATURES

4.1 Site Location

The site for the R.E. Ginna Nuclear Power Plant is located on the south shore of Lake Ontario, approximately 16 miles east of Rochester, New York.

The exclusion area boundary distances from the plant shall be as follows:

<u>Direction</u>	<u>Distance (m)</u>
N (including offshore)	8000
NNE	8000
NE	8000
ENE	8000
E	747
ESE	640
SE	503
SSE	450
S	450
SSW	450
SW	503
WSW	915
W	945
WNW	701
NW	8000
NNW	8000

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircalloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

(continued)

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;
- c. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. However, the consolidated rod storage canister located in Region RGAF2 may exceed these requirements. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be ≤ 2150 BTU/hr.

4.3.1.2 The new fuel storage dry racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;
- c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 257'0" (mean sea level).

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1016 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager, or his designee, shall approve prior to implementation, each proposed test, experiment or modification to structures, systems or components that affect nuclear safety.

5.1.2 The Shift Supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the plant is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the plant is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR;
- b. The plant manager shall report to the corporate vice president specified in 5.2.1.c, shall be responsible for overall safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; and
- c. A corporate vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

5.2 Organization (continued)

5.2.2 Plant Staff

The plant staff organization shall include the following:

- a. An auxiliary operator shall be assigned to the shift crew with fuel in the reactor. An additional auxiliary operator shall be assigned to the shift crew while the plant is in MODE 1, 2, 3 or 4.
 - b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.
 - c. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
 - d. The amount of overtime worked by plant staff members performing safety related functions shall be limited and controlled in accordance with a NRC approved program specified in plant procedures changes to the guidelines in these procedures shall be submitted to the NRC for review.
 - e. The operations manager or operations middle manager shall hold a SRO license.
 - f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant. The STA shall be assigned to the shift crew while the plant is in MODE 1, 2, 3 or 4 and shall meet the qualifications contained in the STA training program specified in UFSAR Section 13.2.
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5.0 ADMINISTRATIVE CONTROLS

5.3 Plant Staff Qualifications

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI Standard N18.1-1971, as supplemented by Regulatory Guide 1.8, Revision 1, September 1975, for comparable positions.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
 - c. Effluent and environmental monitoring;
 - d. Fire Protection Program implementation; and
 - e. All programs specified in Specification 5.5.
-

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),
 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the onsite review function and the approval of the plant manager; and

(continued)

5.5 Programs and Manuals

5.5.1 ODCM (continued)

- c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include Containment Spray, Safety Injection, and Residual Heat Removal in the recirculation configuration. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling Program

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for sampling and analysis; and
- c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I and 40 CFR 141;
- e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)



5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program (continued)

- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;
- h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
- j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in UFSAR Table 5.1-4 to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

(continued)

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g). The program shall include the following:

- a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program.
- b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.
- c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies and methods will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented.

- a. Containment Post-Accident Charcoal System
 1. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
 2. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass $< 1.0\%$, when tested under ambient conditions.
 3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.
- b. Containment Recirculation Fan Cooler System
 1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
 2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass $< 1.0\%$.
- c. Control Room Emergency Air Treatment System (CREATS)
 1. Demonstrate the pressure drop across the HEPA filter bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
 2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass $< 1.0\%$.

(continued)



5.5 Programs and Manuals (continued)

5.5.10 VFTP (continued)

3. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
4. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass $< 1.0\%$, when tested under ambient conditions.
5. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.

d. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.
2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass $< 1.0\%$, when tested under ambient conditions.
3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NUREG-0133.

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and
- b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 1. an API gravity or an absolute specific gravity within limits,
 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

(continued)

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program (continued)

3. a clear and bright appearance with proper color; and
- b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71e.

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the supported system(s) is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.14 SFDP (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 60 psig.

The maximum allowable primary containment leakage rate, L_a , at P_a , shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

- a. Containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) For each air lock, overall leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$, and
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when tested at $\geq P_a$.
- c. Mini-purge valve acceptance criteria is $\leq 0.05 L_a$ when tested at $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling $< 20\%$ of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted on or before April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10.CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)



5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the plant shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

- LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
- LCO 3.1.3, "MODERATOR TEMPERATURE COEFFICIENT (MTC)";
- LCO 3.1.5, "Shutdown Bank Insertion Limit";
- LCO 3.1.6, "Control Bank Insertion Limits";
- LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
- LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)";
- LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
- LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
- LCO 3.9.1, "Boron Concentration."

(continued)



5.6 Reporting Requirements

5.6.5 COLR (continued)

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
(Methodology for LCO 3.1.1, LCO 3.1.3, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.9.1.)
 2. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," Revision 1, February 1982.
(Methodology for LCO 3.2.1.)
 3. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974.
(Methodology for LCO 3.2.3.)
 4. WCAP-8567-P-A, "Improved Thermal Design Procedure," February 1989.
(Methodology for LCO 3.4.1 when using ITDP.)
 5. WCAP 11397-P-A, "Revised Thermal Design Procedure," April 1989.
(Methodology for LCO 3.4.1 when using RTDP.)
 6. WCAP-10054-P-A and WCAP-10081, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
(Methodology for LCO 3.2.1)
 7. WCAP-10924-P-A, Volume 1, Rev. 1, and Addenda 1,2,3, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation," December 1988.
(Methodology for LCO 3.2.1)
 8. WCAP-10924-P-A, Volume 2, Rev. 2, and Addenda, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," December 1988.
(Methodology for LCO 3.2.1)

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

9. WCAP-10924-P-A, Rev. 2 and WCAP-12071, "Westinghouse Large-Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped With Upper Plenum Injection, Addendum 1: Responses to NRC Questions," December 1988.
(Methodology for LCO 3.2.1)
 10. WCAP-10924-P, Volume 1, Rev. 1, Addendum 4, "Westinghouse LBLOCA Best Estimate Methodology; Model Description and Validation; Model Revisions," August 1990.
(Methodology for LCO 3.2.1)
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heatup, cooldown, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
- LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
- b. The power operated relief valve lift settings required to support the Low Temperature Overpressure Protection (LTOP) System, and the LTOP enable temperature shall be established and documented in the PTLR for the following:
- LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves"; and
LCO 3.4.12, "LTOP System."

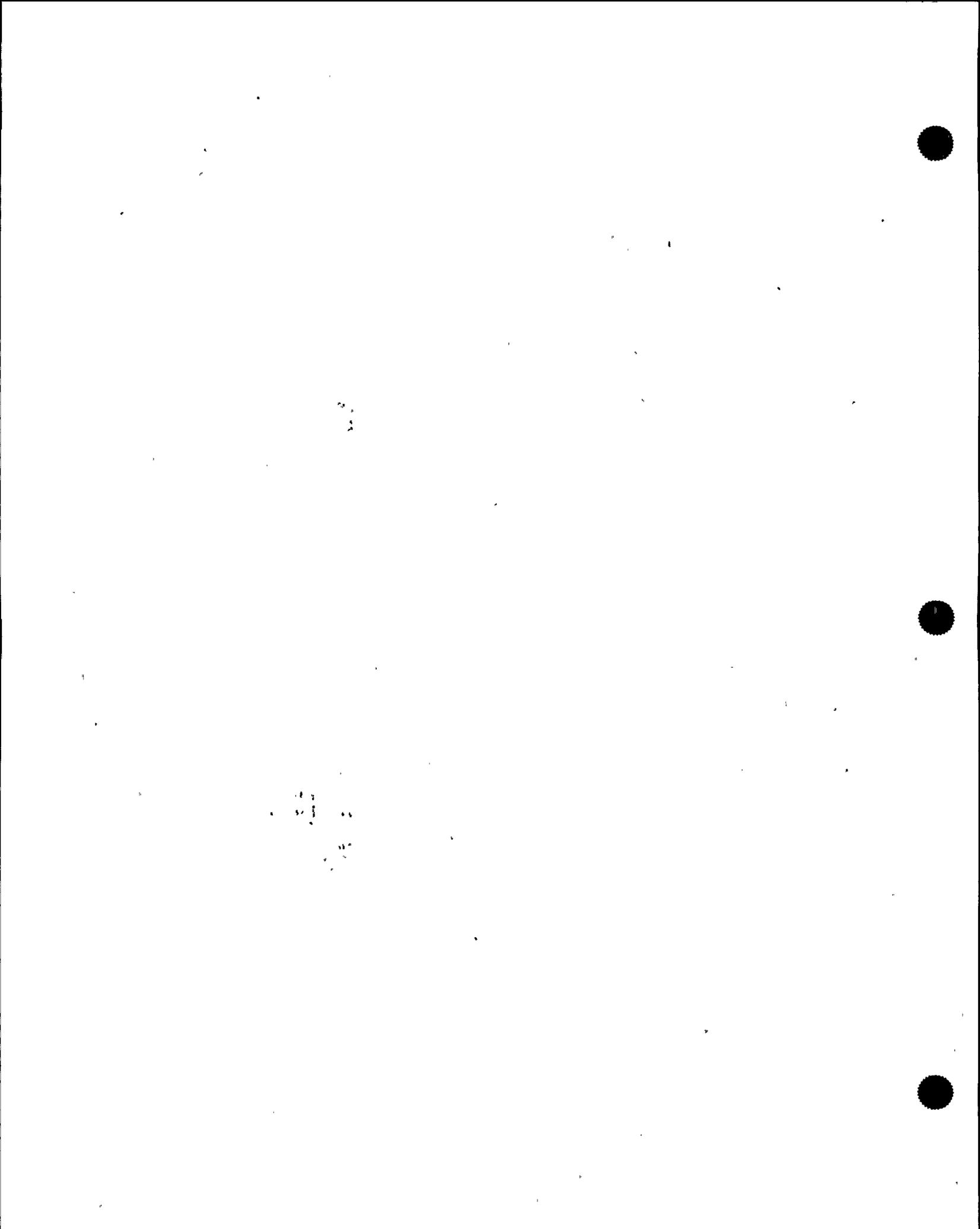
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5.6 Reporting Requirements

5.6.6 PTLR (continued)

- c. The RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC in Amendment No. 48. The acceptability of the P/T and LTOP limits are documented in NRC letter, "R.E. Ginna - Acceptance for Referencing of Pressure Limits Report," December 26, 1995. Specifically, the limits and methodology are described in the following documents:
1. Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant, March 6, 1992.
 2. Letter from C.I. Grimes, NRC, to R.A. Newton, Westinghouse Electric Corporation, "Acceptance for Referencing Topical Report WCAP-14040, Revision 1, 'Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves'," October 16, 1995.
 3. Letter from R.C. Mecredy, Rochester Gas and Electric Corporation (RG&E), to Document Control Desk, NRC, Attention A.R. Johnson, "Technical Specifications Improvement Program, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," December 8, 1995.
- d. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for revisions or supplement thereto.
-



5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(a), in lieu of the requirements of 10 CFR 20.1601(c), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr at a distance of 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

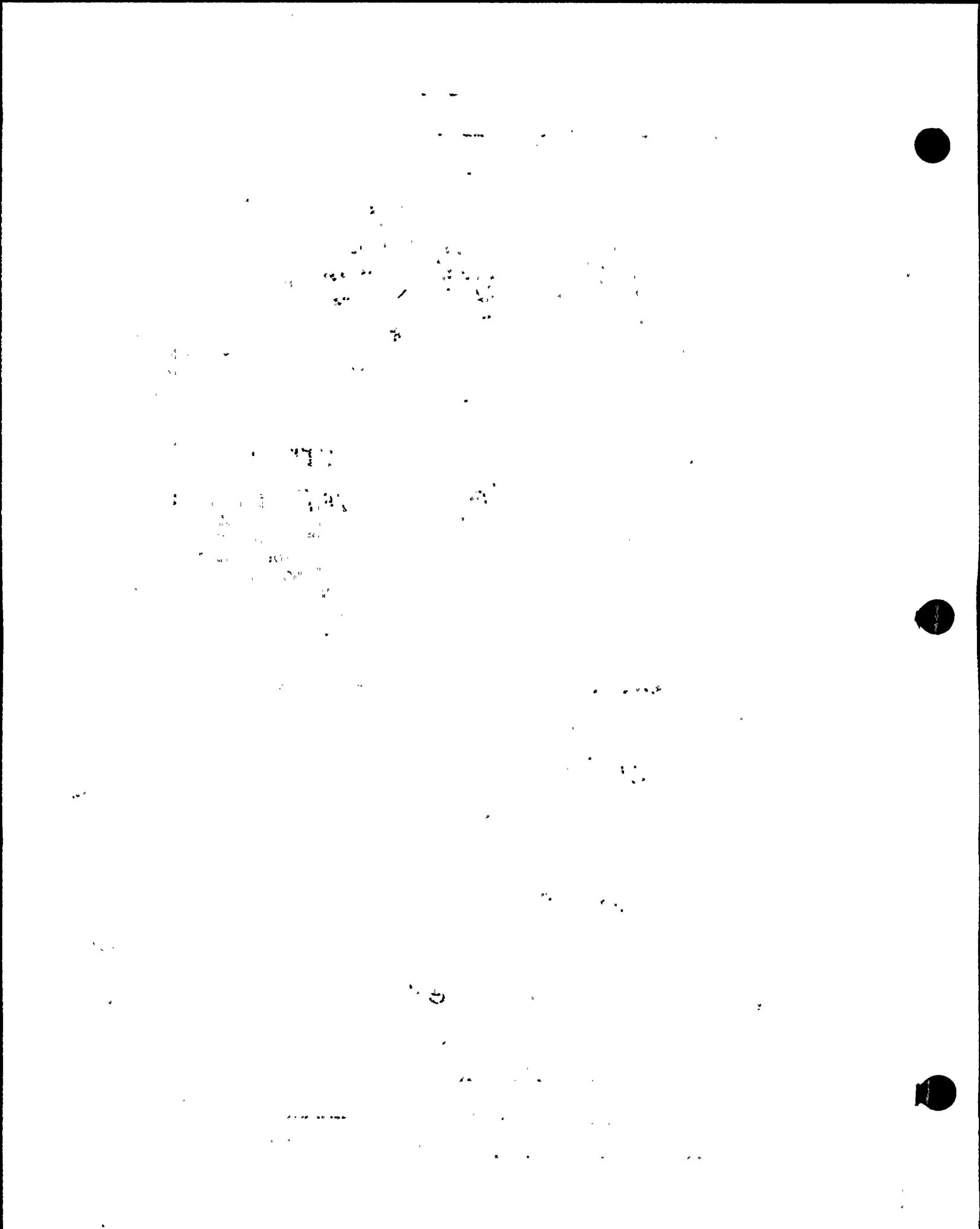
Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.
- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection technician in the RWP.

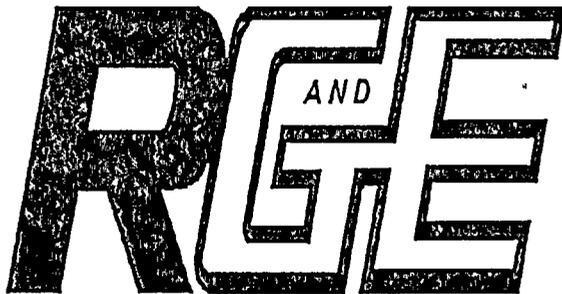
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5.7 High Radiation Area (continued)

- 5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels > 1000 mrem/hr at a distance of 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.
- 5.7.3 In addition to the requirements of Specification 5.7.1, for individual high radiation areas with radiation levels of > 1000 mrem/hr at a distance of 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.
-



50-244
12/28/95
9601030010



Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

Attachment C
Chapters 1.0 - 3.4

Volume II

ATTACHMENT C

Proposed Revised R.E. Ginna Nuclear Power Plant
Improved Technical Specifications

Revise the pages as follows:

Remove

License
Table of Contents
Entire Section 1.0
Entire Section 2.0
Entire Section 3.0
Entire Section 4.0
Entire Section 5.0
Entire Section 6.0

Insert

Ginna Station ITS License
Ginna Station ITS Table of Contents
Ginna Station ITS Section 1.0
Ginna Station ITS Section 2.0
Ginna Station ITS Section 3.0

Ginna Station ITS Section 4.0
Ginna Station ITS Section 5.0

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

FACILITY OPERATING LICENSE

License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;
 - B. Construction of the R. E. Ginna Nuclear Power Plant (the facility) has been substantially completed to conformity with Construction Permit No. CPPR-19, as amended, and the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission (except as exempted from compliance in Section 2.D below);
 - D. There is reasonable assurance (i) that the facility can be operated at power levels up to 1520 megawatts (thermal) without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the regulations of the Commission (except as exempted from compliance in Section 2.D below);
 - E. The applicant is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;
 - F. The applicant has furnished proof of financial protection that satisfies the requirements of 10 CFR Part 140; and
 - G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.
2. The Provisional Operating License dated September 19, 1969, is superseded by Facility Operating License No. DPR-18 hereby issued to Rochester Gas and Electric Corporation to read as follows:



- A. This license applies to the R. E. Ginna Nuclear Power Plant, a closed cycle, pressurized, light-water-moderated and cooled reactor, and electric generating equipment (herein referred to as "the facility") which is owned by the Rochester Gas and Electric Corporation (hereinafter "the licensee" or "RG&E"). The facility is located on the licensee's site on the south shore of Lake Ontario, Wayne County, New York, about 16 miles east of the City of Rochester and is described in license application Amendment No. 6, "Final Facility Description and Safety Analysis Report," and subsequent amendments thereto, and in the application for power increase notarized February 2, 1971, and Amendment Nos. 1 through 4 thereto (herein collectively referred to as "the application").
- B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses RG&E:
- (1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location to Wayne County, New York, in accordance with the procedures and limitations set forth in this license;
 - (2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material or reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as amended, and Commission Safety Evaluations dated November 15, 1976, October 5, 1984, November 14, 1984, and August 30, 1995.
 - (a) Pursuant to the Act and 10 CFR Part 70, to receive and store four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979);
 - (b) Pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979) as supplemented February 20, 1980 and March 5, 1980;
 - (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amount as required;
 - (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis



or instrument calibration or associated with radioactive apparatus or components; and

- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

- (1) Maximum Power Level

RG&E is authorized to operated the facility at steady-state power levels up to a maximum of 1520 megawatts (thermal).

- (2) Technical Specifications

The Technical Specifications contained in Appendix A, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

- (3) Fire Protection

- (a) The licensee shall implement and maintain in effect all fire protection features described in the licensee's submittals referenced in and as approved or modified by the NRC's Fire Protection Safety Evaluation (SE) dated February 14, 1979 and SE supplements dated December 17, 1980, February 6, 1981, June 22, 1981, February 27, 1985 and March 21, 1985 or configurations subsequently approved by the NRC, subject to provision (b) below.

- (b) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

D. Not used.

E. Physical Protection - The licensee shall maintain in effect and fully implement all provisions of the following Commission-approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p), which are being withheld from public disclosure pursuant to 10 CFR 73.21:

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27827 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Robert Emmet Ginna Nuclear Plant Physical Security Plan," with revisions submitted through August 18, 1987; "Robert Emmet Ginna Nuclear Plant Guard Training and Qualification Plan" with revisions submitted through July 30, 1981; and "Robert Emmet Ginna Nuclear Plant Safeguards Contingency Plan" with revisions submitted through April 14, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. This license is effective as of the date of issuance and shall expire at midnight, September 18, 2009.

FROM THE NUCLEAR REGULATORY COMMISSION

Attachment:
Appendix A - Technical Specifications

Date of Issuance:

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1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	<p>A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.</p> <p>The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.</p>

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Revision 1, 1977.

(continued)



1.1 Definitions (continued)

\bar{E} - AVERAGE
DISINTEGRATION ENERGY

\bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half-lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

LEAKAGE

LEAKAGE from the RCS shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

(continued)

1.1 Definitions (continued)

- MODE A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
- OPERABLE - OPERABILITY A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
- PHYSICS TESTS PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:
- a. Described in Chapter 14, Initial Test Program of the UFSAR;
 - b. Authorized under the provisions of 10 CFR 50.59; or
 - c. Otherwise approved by the Nuclear Regulatory Commission (NRC).

(continued)

1.1 Definitions (continued)

PRESSURE AND
TEMPERATURE LIMITS
REPORT (PTLR)

The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.

QUADRANT POWER TILT
RATIO (QPTR)

QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.

RATED THERMAL POWER
(RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1520 Mwt.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and
- b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.

(continued)



1.1 Definitions (continued)

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE
OPERATIONAL TEST
(TADOT)

A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Shutdown	< 0.99	NA	≥ 350
4	Hot Standby ^(b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown ^(b)	< 0.99	NA	≤ 200
6	Refueling ^(c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, or Frequency.

EXAMPLES The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1 LOGICAL CONNECTORS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2 MULTIPLE LOGICAL CONNECTORS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . .	
	<u>OR</u>	
	A.2.1 Verify . . .	
	<u>AND</u>	
	A.2.2.1 Reduce . . .	
	<u>OR</u>	
	A.2.2.2 Perform . . .	
	<u>OR</u>	
	A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the plant is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times

DESCRIPTION
(continued)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. The Completion time extension cannot be used to extend the stated Completion Time for the first inoperable train, subsystem, component, or variable. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

- a. Must exist concurrent with the first inoperability; and
- b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

- a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
- b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

(continued)

1.3 Completion Times

DESCRIPTION
(continued)

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry). An example of a modified "time zero" with the Completion Time expressed as "once per 8 hours" is illustrated in Example 1.3-6, Condition A. In this example, the Completion Time may not be extended.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1 COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-2 DEFAULT CONDITIONS/LCO 3.0.3 ENTRY/COMPLETION TIMES

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One train inoperable.	A.1 Restore train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

When a train is declared inoperable, Condition A is entered. If the train is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable train is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

When a second train is declared inoperable while the first train is still inoperable, Condition A is not re-entered for the second train. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable train. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable trains is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable trains is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

Upon restoring one of the trains to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first train was declared inoperable. This Completion Time may be extended if the train restored to OPERABLE status was the first inoperable train. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second train being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3 MULTIPLE FUNCTION COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. <u>AND</u> One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. <u>OR</u> C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 . (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

(continued)



1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-4 MULTIPLE FUNCTION COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5 SEPARATE ENTRY CONDITION

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific condition, the Note would appear in that Condition, rather than at the top of the ACTIONS table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6 MULTIPLE ACTIONS WITHIN A CONDITION/COMPLETION TIME EXTENSIONS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to $\leq 50\%$ RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered, and the initial performance of Required Action A.1 must be completed within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-7 MULTIPLE ACTIONS WITHIN A CONDITION/COMPLETION
TIME EXTENSIONS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE
COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-1 SINGLE FREQUENCY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the plant is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the plant is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-2 MULTIPLE FREQUENCIES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 1.25 times the stated Frequency extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3 FREQUENCY BASED ON SPECIFIED CONDITION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Required to be performed within 12 hours after \geq 25% RTP. -----</p> <p>Perform channel adjustment.</p>	<p>7 days</p>

The interval continues, whether or not the plant operation is $<$ 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is $<$ 25% RTP, this Note allows 12 hours after power reaches \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was $<$ 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power \geq 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency and the provisions of SR 3.0.3 would apply.



2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained \leq 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.



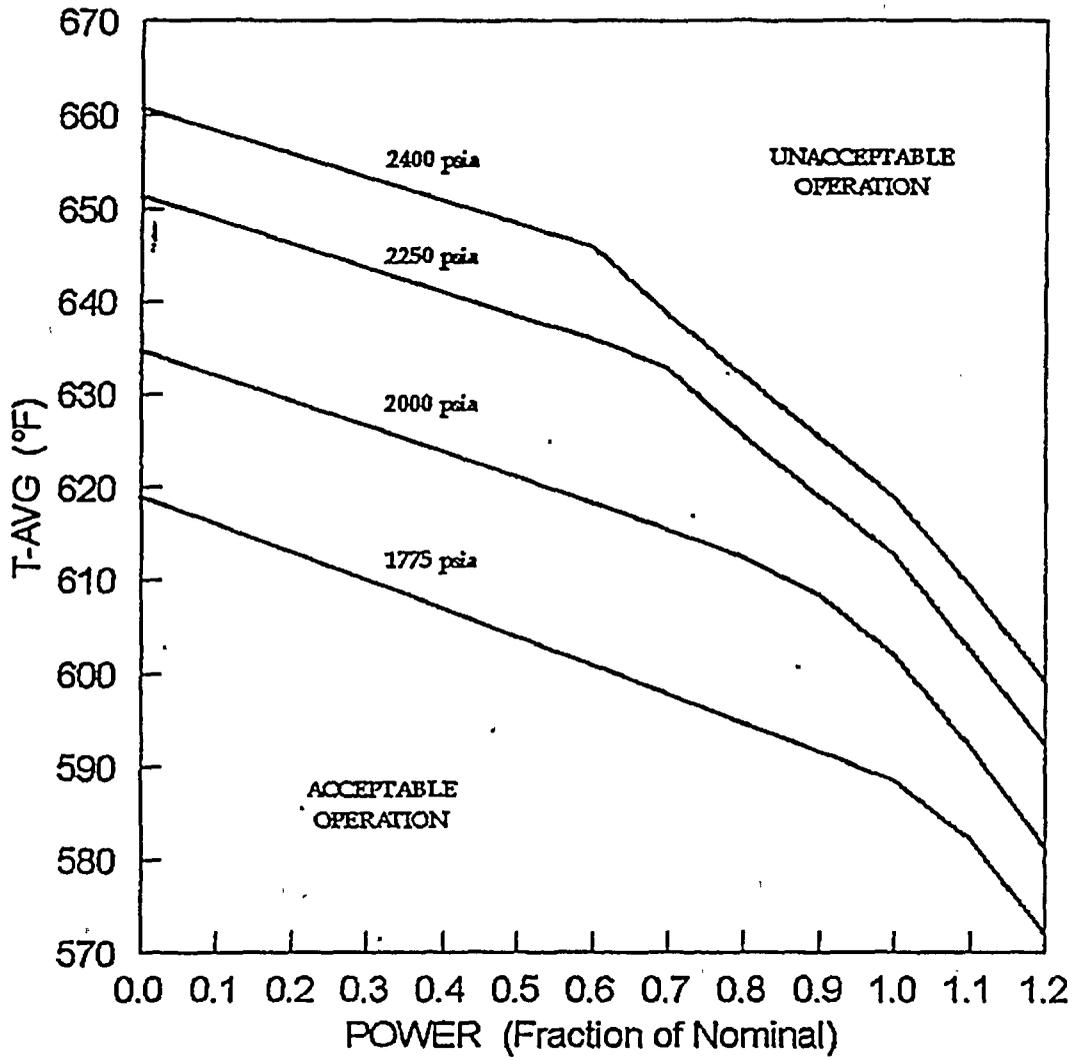


Figure 2.1.1-1
Reactor Safety Limits

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

Atomic Industrial Forum (AIF) GDC 6 (Ref. 1) requires that the reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. This integrity is required during steady state operation, normal operational transients, and anticipated operational occurrences (A00s). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rods and by requiring that fuel centerline temperature stays below the melting temperature (Ref. 2).

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND
(continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium - water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria (Ref. 3):

- a. The hot fuel pellet in the core must not experience centerline fuel melting; and
- b. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit departure from nucleate boiling ratio (DNBR) values that satisfy the DNB design criterion. The observable parameters, thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and/or WRB-1 DNB correlation. These DNB correlations have been developed to predict the DNB flux and the location of DNB for auxiliary uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. A minimum value of the DNB ratio is specified so that during steady state operation, normal operational transients and anticipated transients, there is a 95% probability at a 95% confidence level that DNB will not occur. The curves of Figure 2.1.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which this minimum DNB value is satisfied. The area of safe operation is at or below these lines. Safe operation relative to Figure 2.1.1-1 refers to transient or accident conditions. Normal steady state operation is governed by LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits."

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility (Ref. 4).

The Reactor Trip System setpoints specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation", in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

Automatic enforcement of these reactor core SLs is provided by the following functions (Ref. 5):

- a. High pressurizer pressure trip;
- b. Low pressurizer pressure trip;
- c. Overtemperature ΔT trip;
- d. Overpower ΔT trip;
- e. Power Range Neutron Flux trip; and
- f. Steam generator safety valves.

Additional anticipatory trip functions are also provided for specific abnormal conditions.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (Ref. 6) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

Figure B 2.1.1-1 shows an example of the reactor core safety limits of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is greater than or equal to the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. From this type of figure, the curves on Figure 2.1.1-1 of the accompanying specification can be generated. Each of the curves of Figure 2.1.1-1 has three distinct slopes. Working from left to right, the first slope ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid such that overtemperature ΔT indication remains valid. The second slope ensures that the hot leg steam quality remains $\leq 15\%$ as required by W-3 correlation. The final slope ensures that DNBR is always ≥ 1.3 .

(continued)

BASES

SAFETY LIMITS
(continued)

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT
VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the plant in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage. If the Completion Time is exceeded, actions shall continue in order to bring the plant to a MODE of operation where this SL is not applicable.

(continued)

BASES

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 6, Issued for comment July 10, 1967.
 2. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Deletion of Information Pertaining to Definition of Hot Channel Factors," dated May 30, 1985.
 3. UFSAR, Section 4.2.1.3.3.
 4. UFSAR, Section 4.4.3.
 5. WCAP-8745, "Design Bases for the Thermal Overpower Delta T and Thermal Overtemperature Delta T Trip Functions," March 1977.
 6. UFSAR, Section 7.2.1.1.1.
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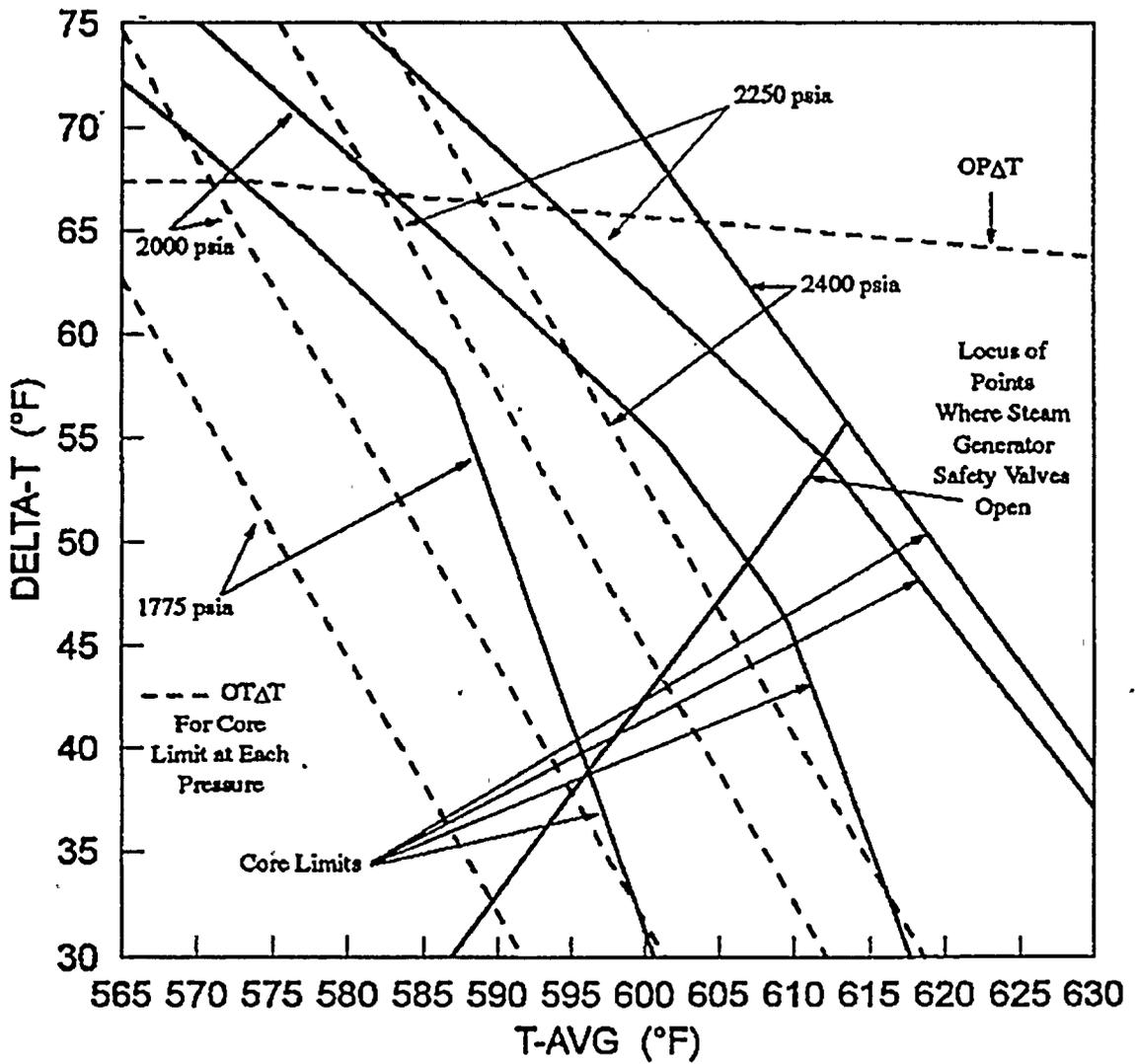


Figure B 2.1.1-1
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to Atomic Industrial Forum (AIF) GDC 9, "Reactor Coolant Pressure Boundary," GDC 33, "Reactor Coolant Pressure Boundary Capability," and GDC 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

The design pressure of the RCS is 2485 psig (Ref. 2). During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 3) except for locked rotor accidents which must be limited to 120% of the design pressure (Refs. 4, 5, and 6). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of plant operation, RCS components are pressure tested, in accordance with the requirements of the approved Ginna ISI/IST Program which is based on ASME Code, Section XI (Ref. 7).

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 8). If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 3) except for locked rotor accidents which must be limited to 120% of the design pressure. The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints (Ref. 9), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization. The safety analyses which credit either the high pressure trip or the RCS pressurizer safety valves are performed using conservative assumptions relative to the other pressure control devices.

More specifically, no credit is taken for operation of the following:

- a. Pressurizer power operated relief valves;
- b. Steam generator atmospheric relief valves;
- c. Steam Dump System;
- d. Reactor Control System;
- e. Pressurizer Level Control System; or
- f. Pressurizer spray valves.

(continued)

BASES

SAFETY LIMITS The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure except for locked rotor accidents which must be limited to 120% of the design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under the original design requirements of USAS B31.1 (Ref. 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS If SL 2.1.2 is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 8).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the Completion Time is exceeded, actions shall continue in order to restore compliance with the SL and bring the plant to MODE 3.

(continued)

BASES

SAFETY LIMIT
VIOLATIONS
(Continued)

If SL 2.1.2 is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. If the Completion Time is exceeded, action shall continue in order to reduce pressure to less than the SL. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 9, 33, and 34, Issued for comment July 10, 1967.
 2. UFSAR, Section 5.1.4.
 3. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
 4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-1, XV-2, XV-3, XV-4, XV-5, XV-6, XV-7, XV-8, XV-10, XV-12, XV-14, XV-15, and XV-17, Design Basis Events, Accidents and Transients (R.E. Ginna)," dated September 4, 1981.
 5. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967 edition.
 6. UFSAR, Section 15.3.2.
 7. ASME, Boiler and Pressure Vessel Code, Section XI.
 8. 10 CFR 100.
 9. UFSAR, Section 7.2.2.2.
-

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and (1) the associated ACTIONS are not met, (2) an associated ACTION is not provided, or (3) if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated to place the plant, as applicable, in:

- a. MODE 3 within 6 hours;
- b. MODE 4 within 12 hours; and
- c. MODE 5 within 36 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to determine OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.7 Test Exception LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a SR, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)



3.0 SR APPLICABILITY

SR 3.0.3
(continued) When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the plant is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

(continued)



BASES

LCO 3.0.2
(continued)

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the plant in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the plant that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. In this instance, the individual LCO's ACTIONS specify the Required Actions. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

(continued)

BASES

LCO 3.0.2
(continued)

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems as required by the LCO. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the plant may enter a MODE or other specified condition in which another Specification becomes applicable and the new LCO is not met. In this case, the Completion Times of the new Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or

(continued)

BASES

LCO 3.0.3
(continued)

- b. The condition of the plant is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the plant. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the plant in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

(continued)

BASES

LCO 3.0.3
(continued)

Upon entering LCO 3.0.3, the Shift Supervisor shall evaluate the condition of the plant and determine actions to be taken, considering plant safety first, that will allow sufficient time for an orderly plant shutdown. These actions shall include preparation for a safe and controlled shutdown, as well as actions to correct the condition which caused entry into LCO 3.0.3. If it is determined that the condition that caused entry into LCO 3.0.3 can be corrected within a reasonable period of time and still allow sufficient time for an orderly plant shutdown, a power reduction does not have to be initiated. This includes coordinating the reduction in electrical generation with energy operations to ensure the stability and availability of the electrical grid. The shutdown shall be initiated so that the time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met.
- b. A Condition exists for which the Required Actions have now been performed.
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

(continued)



BASES

LCO 3.0.3
(continued)

The time limits of LCO 3.0.3 allow 36 hours for the plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 10 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 12 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the plant is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.11, "Spent Fuel Pool (SFP) Water Level." LCO 3.7.11 has an Applicability of "During movement of irradiated fuel assemblies in the SFP." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.11 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the plant in a shutdown condition. The Required Action of LCO 3.7.11 of "Suspend movement of irradiated fuel assemblies in the SFP" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

(continued)

BASES (continued)

LCO 3.0.4 LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the plant in a different MODE or other specified condition stated in the Applicability when the following exist:

- a. Plant conditions are such that the requirements of an LCO would not be met in the MODE or other specified condition in the Applicability desired to be entered; and
- b. The plant would be required to exit the MODE or other specified condition in the Applicability desired to be entered in order to comply with the Required Actions of the affected LCO.

Compliance with Required Actions that permit continued operation of the plant for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the plant before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a shutdown performed in response to the expected failure to comply with ACTIONS.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

(continued)



BASES

LCO 3.0.4
(continued)

LCO 3.0.4 is applicable when entering all MODES, whether increasing in MODES (e.g., MODE 5 to MODE 4) or decreasing in MODES (e.g., MODE 4 to MODE 5). This requirement precluding entry into another MODE when the associated ACTIONS do not provide for continued operation for an unlimited period of time ensures that the plant maintains sufficient equipment OPERABILITY and redundancy as assumed in the accident analyses.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this LCO is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

(continued)

BASES

LCO 3.0.5
(continued)

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support systems' LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

(continued)

BASES

LCO 3.0.6
(continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES (continued)

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the plant. These special tests and operations are necessary to demonstrate select plant performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performances of special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the Test Exception LCO is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

(continued)

BASES

SR 3.0.1
(continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary plant parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

(continued)

BASES

SR 3.0.2
(continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be exceeded by TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with Refueling intervals) or periodic Completion Time intervals beyond those specified.

(continued)

BASES

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified plant conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

BASES

SR 3.0.3
(continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with plant shutdown as well as startup.

The provisions of this specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a shutdown performed in response to the expected failure to comply with ACTIONS.

(continued)



BASES

SR 3.0.4
(continued)

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, train, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is applicable when entering all MODES, whether increasing in MODES (e.g., MODE 5 to MODE 4) or decreasing in MODES (e.g., MODE 4 to MODE 5). This requirement precluding entry into another MODE when the associated ACTIONS do not provide for continued operation for an unlimited period of time ensures that the plant maintains sufficient equipment OPERABILITY and redundancy as assumed in the accident analyses.



3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with $k_{off} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within the limits specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within $\pm 1\% \Delta k/k$ of predicted values.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	<u>AND</u> A:2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.1 -----NOTE----- Required to be performed prior to entering MODE 1. ----- Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p>Once after each refueling</p>
<p>SR 3.1.2.2 -----NOTES----- 1. Only required after 60 effective full power days (EFPD). 2. The predicted reactivity values must be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. ----- Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p>31 EFPD</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be less than or equal to 5 pcm/°F for power levels below 70% RTP and less than or equal to 0 pcm/°F at or above 70% RTP.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Required Action C.1 must be completed whenever Condition C is entered. ----- Projected end of cycle life (EOL) MTC not within lower limit.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. ----- C.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.</p>	<p>Once prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm</p>
<p>D. Required Action and associated Completion Time of Condition C not met.</p>	<p>D.1 Be in MODE 4.</p>	<p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.3.1 Verify MTC is within upper limit.</p>	<p>Once prior to entering MODE 1 after each refueling</p>
<p>SR 3.1.3.2 Confirm that MTC will be within limits at 70% RTP.</p>	<p>Once prior to entering MODE 1 after each refueling</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.3.3 Confirm that MTC will be within limits at EOL.	Once prior to entering MODE 1 after each refueling.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours
B. One rod not within alignment limits.	B.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
		(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce THERMAL POWER to $\leq 75\%$ RTP. <u>AND</u> B.3 Verify SDM is within the limits specified in the COLR. <u>AND</u> B.4 Perform SR 3.2.1.1. <u>AND</u> B.5 Perform SR 3.2.2.1. <u>AND</u> B.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	2 hours Once per 12 hours 72 hours 72 hours 5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 2 with $K_{off} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2 -----NOTE----- Only required to be performed if the rod position deviation monitor is inoperable. ----- Verify individual rod positions within alignment limit.	Once within 4 hours and every 4 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.3 Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core to a MRPI transition in either direction.</p>	<p>92 days</p>
<p>SR 3.1.4.4 Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with:</p> <ul style="list-style-type: none"> a. $T_{avg} \geq 500^{\circ}\text{F}$; and b. Both reactor coolant pumps operating. 	<p>Once prior to reactor criticality after each removal of the reactor head</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limit

LC0 3.1.5 The shutdown bank shall be at or above the insertion limit specified in the COLR.

-----NOTE-----
The shutdown bank may be outside the limit when required for performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
 MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shutdown bank not within limit.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown bank to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify the shutdown bank insertion is within the limit specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

-----NOTE-----
The control bank being tested may be outside the limits when required for the performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank limits not met.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore control bank(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2 Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3 -----NOTE----- Only required to be performed if the rod insertion limit monitor is inoperable. ----- Verify each control bank insertion is within the limits specified in the COLR.	Once within 4 hours and every 4 hours thereafter
SR 3.1.6.4 Verify each control bank not fully withdrawn from the core is within the sequence and overlap limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Microprocessor Rod Position Indication (MRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 with $K_{off} \geq 1.0$.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable MRPI per group and each demand position indicator per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to $\leq 50\%$ RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. More than one MRPI per group inoperable for one or more groups.</p> <p><u>OR</u></p> <p>More than one demand position indicator per bank inoperable for one or more banks.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.7.1 Verify each MRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.</p>	<p>Prior to reactor criticality after each removal of the reactor head</p>



3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

- LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.4, "Rod Group Alignment Limits";
- LCO 3.1.5, "Shutdown Bank Insertion Limit";
- LCO 3.1.6, "Control Bank Insertion Limits";
- LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

- a. THERMAL POWER is maintained \leq 5% RTP;
- b. RCS lowest loop average temperature is \geq 530°F; and
- c. SDM is within the limits specified in the COLR.

APPLICABILITY: During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u> A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Perform a COT on power range and intermediate range channels per SR 3.3.1.7 and SR 3.3.1.8.	Once within 7 days prior to criticality
SR 3.1.8.2 Verify the RCS lowest loop average temperature is $\geq 530^{\circ}\text{F}$.	30 minutes
SR 3.1.8.3 Verify THERMAL POWER is $\leq 5\%$ RTP.	30 minutes
SR 3.1.8.4 Verify SDM is within the limits specified in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 27 and 28 (Ref. 1), two independent reactivity control systems must be available and capable of holding the reactor core subcritical from any hot standby or hot operating condition. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (A00s) which are defined as Condition 2 events in Reference 2 (i.e., events which can be expected to occur during a calendar year with moderate frequency). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperature are changed to the nominal hot zero power temperature.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable rod cluster control assemblies (RCCAs) and soluble boric acid in the Reactor Coolant System (RCS) which each provide a neutron absorbing mechanism. The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The chemical and volume control system can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

(continued)

BASES

BACKGROUND
(continued)

During power operation, SDM control is ensured by operating with the shutdown bank fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank fully withdrawn position is defined in the COLR. When the plant is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE
SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in the safety analyses. The safety analysis (Ref. 3) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out following a scram.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Accidents;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 200 cal/gm energy deposition for the rod ejection accident); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The most limiting accident for the SDM requirements is based on a steam line break (SLB), as described in the accident analysis (Ref. 3). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting SLB for both one loop and two loop operation, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the SLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting SLB transient, the SDM requirement must also protect against:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition;
- c. Startup of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

Each of these events is discussed below.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In the boron dilution analysis (Ref. 4), the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis (i.e., the time available to operators to stop the dilution event). This event is analyzed for refueling, shutdown (MODE 5) and power operation conditions and is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip (Ref. 5). In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits if SDM has been maintained.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core (Ref. 6). The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less severe than the effects of a small steam line break with one loop operation. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition if SDM has been maintained.

The ejection of a control rod constitutes a break in the RCS which rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure (Ref. 7). The ejection of a rod also produces a time dependent redistribution of core power which results in a high neutron flux trip. Fuel and cladding limits are not exceeded if SDM has been maintained.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the plant is operating within the bounds of accident analysis assumptions.

(continued)

BASES (continued)

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration in the RCS.

The COLR provides the shutdown margin requirement with respect to RCS boron concentration. The SLB (Ref. 3) and the boron dilution (Ref. 4) accidents are the most limiting analyses that establish the SDM curve in the COLR. The maximum shutdown margin requirement occurs at end of cycle life and is based on the value used in analysis for the SLB. Early in cycle life, less SDM is required and is bounded by the requirements provided in the COLR. All other accidents analyses are based on 1% reactivity shutdown margin. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 8). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with $k_{\text{eff}} < 1.0$ and in MODES 3, 4 and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODE 1 and MODE 2 with $K_{\text{eff}} \geq 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits."

(continued)

BASES (continued)

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the flowpath of choice would utilize a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 10 gpm using 13,000 ppm boric acid solution, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 10 gpm and 13,000 ppm represent typical values and are provided for the purpose of offering a specific example.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.1.1.1

In MODE 2 with $K_{\text{eff}} < 1.0$ and MODES 3, 4, and 5, the SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27 and 28, Issued for comment July 10, 1967.
 2. "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 3. UFSAR, Section 15.1.5.
 4. UFSAR, Section 15.4.4.
 5. UFSAR, Section 15.4.2.
 6. UFSAR, Section 15.4.3.
 7. UFSAR, Section 15.4.5.
 8. 10 CFR 100.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 30 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN (SDM) or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

(continued)

BASES

BACKGROUND
(continued)

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve) in the core design report, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed or stable (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and normal operating temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant moderator temperature. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the Nuclear Design Methodology provides an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOL, or that an unexpected change in core conditions has occurred.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of $\pm 1\% \Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within $1\% \Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

(continued)

BASES (continued)

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 and MODE 2 with $K_{\text{eff}} \geq 1.0$ because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODE 2 with $K_{\text{eff}} < 1.0$ or MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is only changing because of xenon.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (SR 3.1.2.1).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours. If the SDM for MODE 2 with $K_{eff} < 1.0$ is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison must be made prior to entering MODE 1 when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. Since the reactor must be critical to verify core reactivity, it is acceptable to enter MODE 2 with $K_{eff} \geq 1.0$ to perform this SR. This SR is modified by a Note to clarify that the SR does not need to be performed until prior to entering MODE 1.

SR 3.1.2.2

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Frequency of 31 EFPD, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. The SR is modified by two Notes. The first Note states that the SR is only required after 60 effective full power days (EFPD). The second Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 30, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 8 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). MTC is defined as the change in reactivity per degree change in moderator temperature since temperature is directly proportional to coolant density. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL within the range analyzed in the plant accident analysis. The end of cycle life (EOL) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL limit.

(continued)

BASES

BACKGROUND
(continued)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). Such accidents include the rod withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative (i.e., lower limit). Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is at BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

MTC satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within the specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOL; this upper bound must not be exceeded. This maximum upper limit occurs at BOL, all rods out (ARO), hot zero power (HZP) conditions. At EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check at BOL on MTC provides confirmation that the MTC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOL so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOL positive limit and the EOL negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the plant to take advantage of improved fuel management and changes in plant operating schedule.

If the LCO limits are not met, the plant response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

(continued)



BASES (continued)

APPLICABILITY In MODE 1, the upper and lower limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis since MTC becomes more negative as the cycle burnup increases because the RCS boron concentration is reduced. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

MTC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.

If the upper MTC limit is violated at BOL, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The plant is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

(continued)

BASES

ACTIONS
(continued)

B.1

If the required administrative withdrawal limits at BOL are not established within 24 hours, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status the plant must be brought to MODE 2 with $k_{eff} < 1.0$. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOL MTC lower limit means that the safety analysis assumptions of the EOL accidents that use a bounding negative MTC value may be invalid. If it is determined during physics testing that the EOL MTC value will exceed the most negative MTC limit specified in the COLR, the safety analysis and core design must be re-evaluated prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm to ensure that operation near the EOL remains acceptable. The 300 ppm limit is sufficient to prevent EOL operation at or below the accident analysis MTC assumptions.

Condition C has been modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate at conditions where the MTC would be below the most negative limit specified in the COLR.

Required Action C.1 is modified by a Note which states that LCO 3.0.4 is not applicable. This Note is provided since the requirement to re-evaluate the core design and safety analysis prior to reaching an equivalent RTP ARO boron concentration of 300 ppm is adequate action without restricting entry into MODE 1.

(continued)

BASES

ACTIONS
(continued)

D.1

If the re-evaluation of the accident analysis cannot support the predicted EOL MTC lower limit, or if the Required Actions of Condition C are not completed within the associated Completion Time the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 4 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOL MTC value for ARO will be inferred from isothermal temperature coefficient (ITC) measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

The measurement of the MTC at the beginning of the fuel cycle is adequate to confirm that the MTC remains within its upper limits and will be within limits at 70% RTP, full power and at EOL, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup. This measurement is consistent with the recommendations detailed in Reference 4.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.3.2

This SR requires measurement of MTC at BOL prior to entering MODE 1 after each refueling in order to demonstrate compliance with the 70% RTP MTC limit. The Frequency of "once prior to entering MODE 1 after each refueling" ensures the limit will also be met at higher power levels.

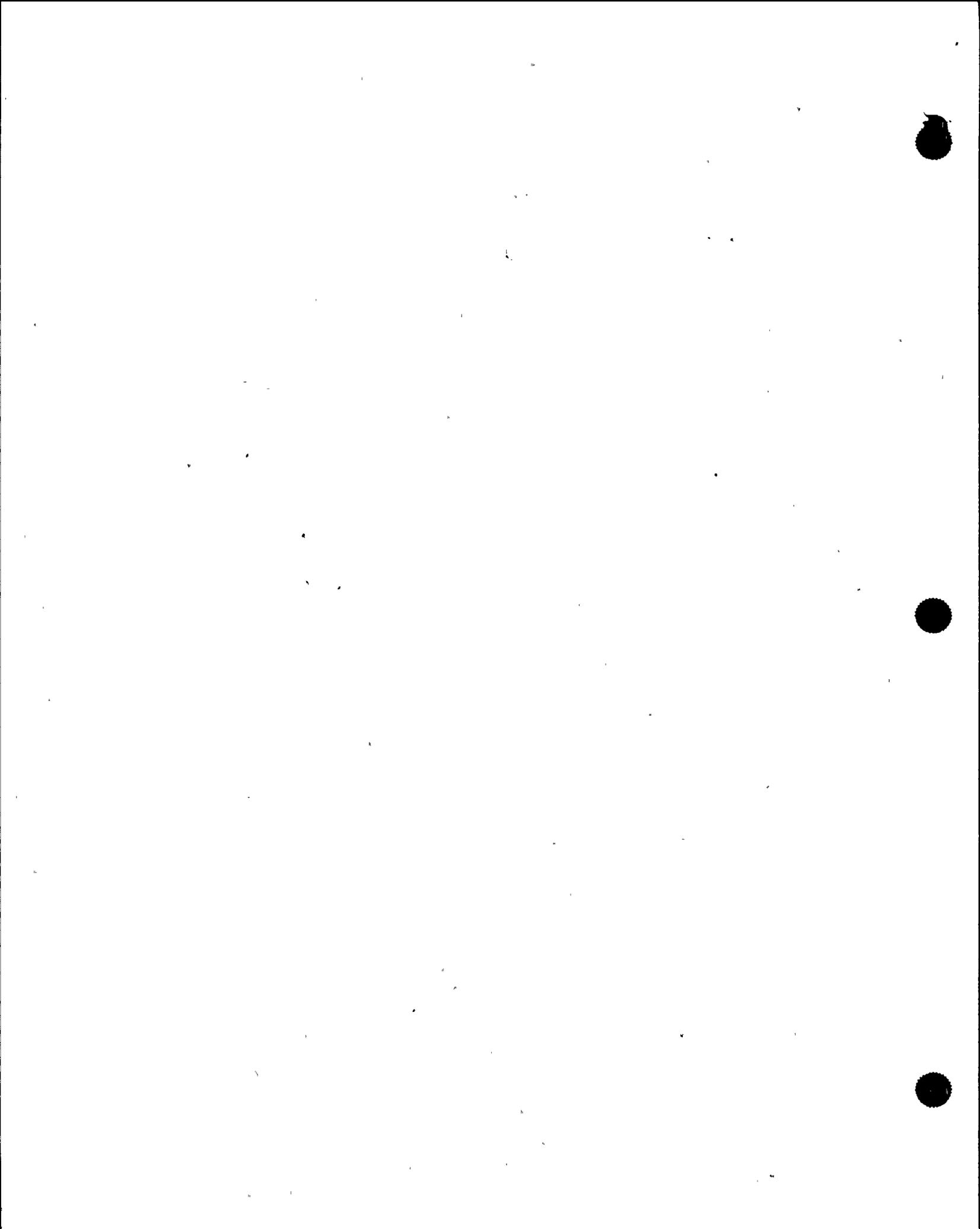
SR 3.1.3.3

This SR requires measurement of MTC at BOL prior to entering MODE 1 after each refueling in order to demonstrate compliance with the most negative MTC LCO. Meeting this limit prior to entering MODE 1 ensures that the limit will also be met at EOL.

The MTC value for EOL is also inferred from the ITC measurements. The EOL value is calculated using the predicted EOL MTC from the core design report and the difference between the measured and predicted ITC. The EOL value is directly compared to the most negative EOL value established in the COLR to ensure that the predicted EOL negative MTC value is within the accident analysis assumptions.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 8, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
 3. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
 4. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06", dated April 28, 1988.
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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

*after each
refueling*

SR 3.1.3.2

This SR requires measurement of MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the 70% RTP MTC limit. The Frequency of "once prior to MODE 1 after each refueling" ensures the limit will also be met at higher power levels. *entering*

SR 3.1.3.3

after each refueling

This SR requires measurement of MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most negative MTC LCO. Meeting this limit prior to entering MODE 1 ensures that the limit will also be met at EOL.

The MTC value for EOL is also inferred from the ITC measurements. The EOL value is calculated using the predicted EOL MTC from the core design report and the difference between the measured and predicted ITC. The EOL value is directly compared to the most negative EOL value established in the COLR to ensure that the predicted EOL negative MTC value is within the accident analysis assumptions.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 8, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
 3. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
 4. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06", dated April 28, 1988.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 6, 14, 27 and 28 (Ref. 1), and 10 CFR 50.46 (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are movable neutron absorbing devices which are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately $\frac{5}{8}$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(continued)

BASES

BACKGROUND
(continued)

The RCCAs are divided among control banks and a shutdown bank. Control banks are used to compensate for changes in reactivity due to variations in operating conditions of the reactor such as coolant temperature, power level, boron or xenon concentration. The shutdown bank provides additional shutdown reactivity such that the total shutdown worth of the bank is adequate to provide shutdown for all operating and hot zero power conditions with the single RCCA of highest reactivity worth fully withdrawn. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station.

The shutdown bank is maintained either in the fully inserted or fully withdrawn position. The fully withdrawn position is defined in the COLR. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Microprocessor Rod Position Indication (MRPI) System.

(continued)

BASES

BACKGROUND
(continued)

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{1}{8}$ inch), but if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The MRPI System also provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The MRPI system consists of one digital detector assembly per rod. All the detector assemblies consist of one coil stack which is multiplexed and becomes input to two redundant MRPI signal processors. Each signal processor independently monitors all rods and senses a rod bottom for any rod. The MRPI system directly senses rod position in intervals of 12 steps for each rod. The digital detector assemblies consist of 20 discrete coil pairs spaced at 12-step intervals. The true rod position is always within ± 8 steps of the indicated position (± 6 steps due to the 12-step interval and ± 2 steps transition uncertainty due to processing and coil sensitivity). With an indicated deviation of 12 steps between the group step counter and MRPI, the maximum deviation between actual rod position and the demand position would be 20 steps, or 12.5 inches.

The safety concerns associated with the MRPI system are associated with generation of a rod drop/rod stop signal which blocks auto rod withdrawal and the ability to comply with the rod misalignment requirement. A rod bottom signal from both signal processors is required to generate a rod drop/rod stop signal. The two-out-of-two coincident signal requirement reduces inadvertent rod drop/rod stop but does not affect the accident analysis assumptions.

The bank demand position and the MRPI rod position signals are monitored by a rod deviation monitoring system that provides an alarm whenever the individual rod position signal deviates from the bank demand signal by > 12 steps. The rod deviation alarm will be generated by the Plant Process Computer System (PPCS).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue (i.e., static rod misalignment). This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Three types of analysis are performed in regard to static rod misalignment (Ref. 4). The first type of analysis considers the case where any one rod is completely inserted into the core with all other rods completely withdrawn. With control banks at their insertion limits, the second type of analysis considers the case when any one rod is completely inserted into the core. The third type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in all three of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

The second type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA fully withdrawn following a main steam line break (Ref. 5).

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F_{\Delta H}^N$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

All shutdown and control rods must be OPERABLE to provide the negative reactivity necessary to provide adequate shutdown for all operating and hot zero power conditions. Shutdown and control rod OPERABILITY is defined as being trippable such that the necessary negative reactivity assumed in the accident analysis is available. If a control rod(s) is discovered to be immovable but remains trippable and aligned, the control rod is considered to be OPERABLE.

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

(continued)

BASES

LCO
(continued)

The requirement to maintain the rod alignment of each individual rod position as indicated by MRPI to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis with respect to power distribution and SDM is 25 steps, while a total misalignment from fully withdrawn to fully inserted is assumed for the control rod misalignment accident.

The rod position deviation monitor is used to verify rod alignment on a continuous basis and will provide an alarm whenever the individual rod position signal deviates from the bank demand signal by > 12 steps. Verification that the rod positions are within the alignment limit is made every 12 hours (SR 3.1.4.1). When the rod position deviation monitor is inoperable a verification that the rod positions are within limit must be made more frequently (SR 3.1.4.2).

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

(continued)

BASES (continued)

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM. Boration is assumed to continue until the required SDM is restored.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a remaining rod of maximum worth.

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{off} < 1.0$ within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{off} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

When a rod is misaligned, it can usually be moved and is still trippable. If the rod cannot be realigned within 1 hour, then SDM must be verified to be within the limits specified in the COLR or boration must be initiated to restore the SDM. The Completion Time of 1 hour gives the operator sufficient time to perform either action in an orderly manner.

(continued)

BASES

ACTIONS
(continued)

B.2, B.3, B.4, B.5, and B.6

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to $\leq 75\%$ RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 6). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1 and SR 3.2.2.1) ensures that current operation at $\leq 75\%$ RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS
(continued)

C.1

When Required Actions of Condition B cannot be completed within their Completion Time, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{\text{eff}} < 1.0$ within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 2 with $K_{\text{eff}} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration is assumed to continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the plant conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{\text{eff}} < 1.0$ within 6 hours.

(continued)

BASES

ACTIONS

D.2 (continued)

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{off} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits using MRPI or the PPCS at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

When the rod position deviation monitor (i.e., the PPCS) is inoperable, no control room alarm is available between the normal 12 hour Frequency to alert the operators of a rod misalignment. A reduction of the Frequency to 4 hours provides sufficient monitoring of the rod positions when the monitor is inoperable. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

This SR is modified by a Note that states that performance of this SR is only necessary when the rod position deviation monitor is inoperable.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.4.3

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 with $K_{off} \geq 1.0$, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod to a MRPI transition will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. During or between required performances of SR 3.1.4.3 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.4

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with both RCPs operating and the average moderator temperature $\geq 500^{\circ}\text{F}$ to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

(continued)



BASES (continued)

- REFERENCES
1. Atomic Industrial Forum (AIF) GDC 6, 14, 27, and 28, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.4.6.
 5. UFSAR, Section 15.1.5.
 6. UFSAR, Section 15.4.2.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limit

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and a shutdown bank. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The shutdown bank insertion limit is defined in the COLR. The shutdown bank is required to be at or above the insertion limit lines.

(continued)

BASES

BACKGROUND
(continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity change associated with large changes in RCS temperature.

The design calculations are performed with the assumption that the shutdown bank is withdrawn first. The shutdown bank can be fully withdrawn without the core going critical. The fully withdrawn position is defined in the COLR. This provides available negative reactivity in the event of boration errors. The shutdown bank is controlled manually by the control room operator. The shutdown bank is either fully withdrawn or fully inserted. The shutdown bank must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown bank is then left in this position until the reactor is shut down. The shutdown bank affects core power and burnup distribution, and adds negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

(continued)

BASES

BACKGROUND
(continued)

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the shutdown and control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE
SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown bank and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown bank shall be at or above the insertion limit and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and the shutdown bank (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment is that:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. RCS pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limit affects safety analysis involving core reactivity and SDM (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Refs. 4, 5, 6, and 7).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits, together with AFD, QPTR and the control and shutdown bank alignment limits, ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Refs. 4, 5, 6, and 7).

The shutdown bank insertion limit preserves an initial condition assumed in the safety analyses and, as such, satisfies Criterion 2 of the NRC Policy Statement.

LCO

The shutdown bank must be at or above the insertion limit any time the reactor is critical and prior to withdrawal of any control rod. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

(continued)

BASES

LCO
(continued)

The LCO is modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

The shutdown bank insertion limit is defined in the COLR.

APPLICABILITY

The shutdown bank must be within the insertion limit, with the reactor in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 2 with $K_{eff} < 1.0$ and MODE 3, 4, 5, or 6, the shutdown bank insertion limit does not apply because the reactor is shutdown and not producing fission power. In shutdown MODES the OPERABILITY of the shutdown rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. Refer to LCO 3.1.1 for SDM requirements in MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

A.1.1, A.1.2, and A.2

When the shutdown bank is not within insertion limit, verification of SDM or initiation of boration to regain SDM within 1 hour is required, since the SDM in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). If the shutdown bank is not within the insertion limit, then SDM will be verified by performing a reactivity balance calculation, taking into account RCS boron concentration, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. Two hours is allowed to restore the shutdown bank to within the insertion limit. This time limit is necessary because the available SDM may be significantly reduced, with the shutdown bank not within the insertion limit. The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to a MODE where the LCO is not applicable. To achieve this status, the plant must be placed in MODE 2 with $k_{eff} < 1.0$ within a Completion Time of 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1

Since the shutdown bank is positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of every 12 hours is adequate to ensure that the bank is within the insertion limit. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

(continued)

BASES (continued)

- REFERENCES
1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.1.5.
 5. UFSAR, Section 15.4.1.
 6. UFSAR, Section 15.4.2.
 7. UFSAR, Section 15.4.6.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and a shutdown bank. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The insertion limits figure in the COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks.

(continued)

BASES

BACKGROUND
(continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. The fully withdrawn position is defined in the COLR. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature.

The rod insertion limit monitor is used to verify control rod insertion on a continuous basis and will provide an alarm whenever the control bank insertion deviates from the rod insertion limits specified in the COLR. Verification that the control banks are within the insertion limit is made every 12 hours (SR 3.1.6.2). When the rod insertion limit monitor is inoperable a verification that the rod positions are within the limit must be made more frequently (SR 3.1.6.3).

The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

(continued)

BASES

BACKGROUND
(continued)

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the shutdown and control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and ensure the required SDM is maintained.

Operation within the AFD, QPTR, shutdown and control bank insertion and alignment LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown bank and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown bank shall be at or above the insertion limit and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and the shutdown bank (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The control bank insertion limits also limit the reactivity worth of an ejected control bank rod.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
 - 1. Specified acceptable fuel design limits, or
 - 2. Reactor Coolant System pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the control bank insertion limits affect safety analysis involving core reactivity and power distributions (Refs. 4, 5, 6, and 7).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Refs. 4, 5, 6, and 7).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits, together with AFD, QPTR and the control and shutdown bank alignment limits, ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Refs. 4, 5, 6, and 7).

The control bank insertion, sequence and overlap limits satisfy Criterion 2 of the NRC Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is limited, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

The rod insertion limit monitor is used to verify control rod insertion on a continuous basis and will provide an alarm whenever the control bank insertion deviates from the rod insertion limits specified in the COLR. Verification that the control banks are within the insertion limit is made every 12 hours (SR 3.1.6.2). When the rod insertion limit monitor is inoperable a verification that the rod positions are within the limit must be made more frequently (SR 3.1.6.3).

The LCO is modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.3. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

(continued)

BASES (continued)

APPLICABILITY The control bank insertion, sequence, and overlap limits shall be maintained with the reactor in MODE 1 and MODE 2 with $k_{\text{eff}} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with $k_{\text{eff}} < 1.0$ and MODES 3, 4, 5, and 6 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

ACTIONS A.1.1, A.1.2, and A.2

When the control banks are outside the acceptable insertion limits, out of sequence, or in the wrong overlap configuration, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position; or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM within 1 hour is required, since the SDM in MODES 1 and 2 is normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). If control banks are not within their limits, then SDM will be verified by performing a reactivity balance calculation, taking into account RCS boron concentration, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. Thus, the allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $K_{eff} < 1.0$, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. The Frequency of within 4 hours prior to achieving criticality ensures that the estimated control bank position is within the limits specified in the COLR shortly before criticality is reached.

SR 3.1.6.2

With an OPERABLE bank insertion limit monitor (i.e., the PPCS), verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.6.3

When the insertion limit monitor (i.e., the PPCS) becomes inoperable, no control room alarm is available between the normal 12 hour frequency to alert the operators of a control bank not within the insertion limits. A reduction of the Frequency to every 4 hours provides sufficient monitoring of control rod insertion when the monitor is inoperable. Verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

This SR is modified by a Note that states that performance of this SR is only necessary when the rod insertion limit monitor is inoperable.

SR 3.1.6.4

When control banks are maintained within their insertion limits as required by SR 3.1.6.2 and SR 3.1.6.3 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.1.5.
 5. UFSAR, Section 15.4.1.
 6. UFSAR, Section 15.4.2..
 7. UFSAR, Section 15.4.6.
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B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

The OPERABILITY (i.e., trippability), including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). Rod position indication is required to assess OPERABILITY and misalignment.

According to the Atomic Industrial Forum (AIF) GDC 12 and 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are movable neutron absorbing devices which are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(continued)

BASES

BACKGROUND
(continued)

The RCCAs are divided among control banks and a shutdown bank. Control banks are used to compensate for changes in reactivity due to variations in operating conditions of the reactor such as coolant temperature, power level, boron or xenon concentration. The shutdown bank provides additional shutdown reactivity such that the total shutdown worth of the bank is adequate to provide shutdown for all operating and hot zero power conditions with the single RCCA of highest reactivity worth fully withdrawn. Each bank is further subdivided into groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Microprocessor Rod Position Indication (MRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{5}{8}$ inch), but if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

(continued)

BASES

BACKGROUND
(continued)

The MRPI System also provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The MRPI system consists of one digital detector assembly per rod. All the detector assemblies consist of one coil stack which is multiplexed and becomes input to two redundant MRPI signal processors. Each signal processor independently monitors all rods and senses a rod bottom for any rod. The MRPI system directly senses rod position in intervals of 12 steps for each rod. The digital detector assemblies consist of 20 discrete coil pairs spaced at 12-step intervals. The true rod position is always within ± 8 steps of the indicated position (± 6 steps due to the 12-step interval and ± 2 steps transition uncertainty due to processing and coil sensitivity). With an indicated deviation of 12 steps between the group step counter and MRPI, the maximum deviation between actual rod position and the demand position would be 20 steps, or 12.5 inches.

APPLICABLE
SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth limits, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

(continued)

BASES (continued)

LCO

LCO 3.1.7 specifies that the MRPI System and the Bank Demand Position Indication System be OPERABLE. For the control rod position indicators to be OPERABLE requires the following:

- a. For the MRPI System there are no failed coils and rod position indication is available on the MRPI screen (in either the control room or relay room) or the plant process computer system; and
- b. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the MRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the MRPI System as required by SR 3.1.7.1 indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of control rod bank position. A deviation of less than the allowable 12 step agreement limit, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis.

The MRPI system is designed with error detection such that when a fault occurs in the binary data received from the coil stacks or processing unit an alarm is annunciated at the MRPI display. When the fault clears, the system provides self validation of data integrity and returns to its normal display mode. Because of the digital nature of the system and its inherent diagnostic features, intermittent data alarms can mask position indication and generate the perception that a single rod position is unmonitored. For a single rod position indication failure, MRPI is considered OPERABLE if a fault occurs and clears within five minutes and the indicated position is within expected values.

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

(continued)

BASES (continued)

APPLICABILITY The requirements on the MRPI and step counters are only applicable in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ (consistent with LCO 3.1.4 and LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which the reactor is critical, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODE 2 with $K_{eff} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable MRPI per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one MRPI per group fails, the position of the rod can still be determined by use of the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

ACTIONS
(continued)

A.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

When one or more rods with inoperable position indicators (i.e., MRPI) have been moved > 24 steps in one direction since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

Acceptable verification of rod position within 4 hours re-initiates the clock for Required Action A.1.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP, if one or more rods are misaligned by more than 24 steps.

(continued)

BASES

ACTIONS
(continued)

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the MRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps from the OPERABLE demand position indicator for that bank within the allowed Completion Time of once every 8 hours is adequate. This ensures that most withdrawn and least withdrawn rod are no more than 24 steps apart which is less than the accident analysis assumption of 25 steps. This verification can be an examination of logs, administrative controls, or other information that shows that all MRPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position will not cause core peaking to approach the core peaking factor limits.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

E.1

With more than one MRPI per group inoperable for one or more groups or more than one demand position indicator per bank inoperable for one or more banks, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1

Verification that the MRPI agrees with the group demand position within 12 steps for the full indicated range of rod travel ensures that the MRPI is operating correctly. Since the MRPI does not display the actual shutdown rod positions between 0 and 230 steps, only points within the indicated ranges are required in comparison.

This Surveillance is performed during a plant outage or during plant startup, prior to reactor criticality after each removal of the reactor head due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 12 and 13, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in the design and analysis;
- c. Verify the assumptions used to predict plant response;
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power, power ascension, and at power operation; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

(continued)

BASES

BACKGROUND
(continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS performed at Ginna Station for reload fuel cycles in MODE 2 include:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and
- d. Isothermal Temperature Coefficient (ITC).

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance as described below.

- a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, bank D is at or near its fully withdrawn position. HZP is where the core is critical ($k_{\text{eff}} = 1.0$), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test could violate LCO 3.1.3, "Moderator Temperature Coefficient (MTC)."

(continued)



BASES

BACKGROUND
 (continued)

- b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ fully inserted into the core. This test is used to measure the differential boron worth. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The differential boron worth is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits;" LCO 3.1.5, "Shutdown Bank Insertion Limit;" or LCO 3.1.6, "Control Bank Insertion Limits."
- c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has two alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

(continued)

BASES

BACKGROUND
(continued)

- d. The ITC Test measures the ITC of the reactor. This test is performed at HZP using the Slope Method. The Slope Method varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The Moderator Temperature Coefficient (MTC) at BOL, 70% RTP and at EOL is determined from the measured ITC. This test satisfies the requirements of SR 3.1.3.1 and SR 3.1.3.2. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."
-

APPLICABLE
SAFETY ANALYSES

The fuel is protected by multiple LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of these LCOs, that are excepted by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 3). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Reference 4 summarizes the initial zero, low power, and power tests. Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines and established industry practices which are consistent with the PHYSICS TESTS described in References 5 and 6. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. The requirements specified in the following LCOs may be suspended for PHYSICS TESTING:

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limit";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality".

When these LCOs are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to $\leq 5\%$ RTP, the reactor coolant temperature is kept $\geq 530^\circ\text{F}$, and SDM is within the limits specified in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the plant safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits to conduct PHYSICS TESTS in MODE 2, to verify certain core physics parameters. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

- a. THERMAL POWER is maintained $\leq 5\%$ RTP;
 - b. RCS lowest loop average temperature is $\geq 530^\circ\text{F}$; and
 - c. SDM is within the limits specified in the COLR.
-

(continued)



BASES (continued)

APPLICABILITY This LCO is applicable when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits since a MODE change has occurred. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS loop with the lowest T_{avg} is < 530°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 530°F could violate the assumptions for accidents analyzed in the safety analyses.

(continued)

BASES

ACTIONS
(continued)

D.1

If Required Action C.1 cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from MODE 2 in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 7 days prior to criticality. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 7 day time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is $\geq 530^{\circ}\text{F}$ will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.8.3

Verification that THERMAL POWER is < 5% RTP using the NIS detectors will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

The SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
 4. UFSAR, Section 14.6.
 5. Letter from R. W. Kober (RGE) to T. E. Murley (NRC), Subject: "Startup Reports," dated July 9, 1984.
 6. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06," dated April 28, 1988.
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3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor (F_Q(Z))

LCO 3.2.1 F_Q(Z) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F _Q (Z) not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% F _Q (Z) exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce AFD acceptable operation limits ≥ 1% for each 1% F _Q (Z) exceeds limit.	8 hours
	<u>AND</u>	
	A.3 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% F _Q (Z) exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Reduce Overpower ΔT and Overtemperature ΔT trip setpoints ≥ 1% for each 1% F _Q (Z) exceeds limit.	72 hours
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 Perform SR 3.2.1.1 or SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify measured values of F _a (Z) are within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

(continued)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.2</p> <p>-----NOTE----- Only required to be performed if one power range channel is inoperable with THERMAL POWER \geq 75% RTP. -----</p> <p>Verify measured values of F_a(Z) are within limits specified in the COLR.</p>	<p>Once within 24 hours and every 24 hours thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

 LCO 3.2.2 $F_{\Delta H}^N$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_{\Delta H}^N$ not within limit.	A.1 Reduce THERMAL POWER $\geq 1\%$ RTP for each 1% $F_{\Delta H}^N$ exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux-High trip setpoints $\geq 1\%$ for each 1% $F_{\Delta H}^N$ exceeds limit.	72 hours
	<u>AND</u>	
	A.3 Reduce Overpower ΔT and Overtemperature ΔT trip setpoints $\geq 1\%$ for each 1% $F_{\Delta H}^N$ exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Perform SR 3.2.2.1 or SR 3.2.2.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify $F_{\Delta H}^N$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter
SR 3.2.2.2 -----NOTE----- Only required to be performed if one power range channel is inoperable with THERMAL POWER \geq 75% RTP. ----- Verify $F_{\Delta H}^N$ is within limits specified in the COLR.	Once within 24 hours and every 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD monitor alarm shall be OPERABLE and AFD:

- a. Shall be maintained within the target band about the target flux difference with THERMAL POWER \geq 90% RTP. The target band is specified in the COLR.
- b. May deviate outside the target band with THERMAL POWER $<$ 90% RTP but \geq 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is \leq 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.
- c. May deviate outside the target band with THERMAL POWER $<$ 50% RTP.

-----NOTES-----

1. The AFD shall be considered outside the target band when the average of four OPERABLE excore channels indicate AFD to be outside the target band. If one excore detector is out of service, the remaining three detectors shall be used to derive the average.
 2. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with THERMAL POWER \geq 50% RTP, and AFD outside the target band.
 3. Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with THERMAL POWER $>$ 15% RTP and $<$ 50% RTP, and AFD outside the target band.
 4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6.
-

APPLICABILITY: MODE 1 with THERMAL POWER $>$ 15% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. THERMAL POWER \geq 90% RTP.</p> <p><u>AND</u></p> <p>AFD not within the target band.</p>	<p>A.1 Restore AFD to within target band.</p>	<p>15 minutes</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Reduce THERMAL POWER to < 90% RTP.</p>	<p>15 minutes</p>
<p>C. THERMAL POWER < 90% RTP and \geq 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours.</p> <p><u>OR</u></p> <p>THERMAL POWER < 90% RTP and \geq 50% RTP with AFD not within the target band and not within the acceptable operation limits.</p>	<p>C.1 Reduce THERMAL POWER to < 50% RTP.</p>	<p>30 minutes</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. THERMAL POWER \geq 90% RTP.</p> <p><u>AND</u></p> <p>AFD monitor alarm inoperable.</p>	<p>D.1 Perform SR 3.2.3.1.</p>	<p>Once every 15 minutes</p>
<p>E. THERMAL POWER < 90% RTP.</p> <p><u>AND</u></p> <p>AFD monitor alarm inoperable.</p>	<p>E.1 Perform SR 3.2.3.2.</p>	<p>Once every 1 hour</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD monitor is OPERABLE.	12 hours
SR 3.2.3.2 -----NOTES----- 1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER \geq 90% RTP. 2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available. ----- Verify AFD is within limits and log AFD for each OPERABLE excore channel.	Once within 15 minutes and every 15 minutes thereafter
SR 3.2.3.3 -----NOTES----- 1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER < 90% RTP. 2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available. ----- Verify AFD is within limits and log AFD for each OPERABLE excore channel.	Once within 1 hour and every 1 hour thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.2.3.4 Update target flux difference.</p>	<p>Once within 31 EFPD after each refueling</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>
<p>SR 3.2.3.5 -----NOTE----- The initial target flux difference after each refueling may be determined from design predictions. -----</p> <p>Determine, by measurement, the target flux difference.</p>	<p>Once within 31 EFPD after each refueling</p> <p><u>AND</u></p> <p>92 EFPD thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR monitor alarm shall be OPERABLE and QPTR shall be ≤ 1.02 .

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Limit THERMAL POWER to $\geq 3\%$ below RTP for each 1% of QPTR > 1.00.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.4.1 and limit THERMAL POWER to $\geq 3\%$ below RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>	
	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours
		<u>AND</u>
		Once per 7 days thereafter
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.6</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Only required to be performed if the cause of the QPTR alarm is not associated with inoperable QPTR instrumentation. 2. Required Action A.6 must be completed when Required Action A.5 is completed and Note 1, above, does not apply. 3. Only one of the Completion Times, whichever becomes applicable first, must be met. <p>-----</p> <p>Perform SR 3.2.1.1 and SR 3.2.2.1.</p>	<p>Within 24 hours after reaching RTP</p> <p><u>OR</u></p> <p>Within 48 hours after increasing THERMAL POWER increased above the limits of Required Actions A.1 and A.2</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to \leq 50% RTP.	4 hours
C. QPTR monitor alarm inoperable.	C.1 Perform SR 3.2.4.2	Once within 24 hours and every 24 hours thereafter
	<u>OR</u> C.2 Perform SR 3.2.1.2 and SR 3.2.2.2	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.4.1 Verify QPTR monitor alarm is OPERABLE.	12 hours
SR 3.2.4.2 -----NOTES----- 1. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. With one power range channel inoperable and THERMAL POWER \geq 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2. ----- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.3 -----NOTES----- 1. Only required to be performed if the QPTR monitor alarm is inoperable. 2. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 3. With one power range channel inoperable and THERMAL POWER \geq 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2. ----- Verify QPTR is within limit by calculation.	Once within 24 hours and every 24 hours thereafter

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height of the core (Z).

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions adjusted for uncertainty. Therefore, $F_Q(Z)$ is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis.

$F_Q(Z)$ is sensitive to fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using the incore detector system. Measurements are generally taken with the core at or near steady state conditions. With the measured three dimensional power distributions, it is possible to determine a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include variations in the value of $F_Q(Z)$, which are present during a nonequilibrium situation such as load following when the plant changes power level to match grid demand peaks and valleys.

Core monitoring and control under transient conditions (i.e., Condition 1 events as described in Reference 1) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion, Sequence and Overlap Limits.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Limits on F_Q(Z) preclude core power distributions that violate the following fuel design criteria:

- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 2);
- c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 3); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN (SDM) with the highest worth control rod stuck fully withdrawn (Ref. 4).

Limits on F_Q(Z) ensure that the value of the total peaking factor assumed as an initial condition in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

The F_Q(Z) limits provided in the COLR are based on the limits used in the LOCA analysis. F_Q(Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F_Q(Z) assumed in safety analyses for other accidents because of the requirements set forth in 10 CFR 50.46 (Ref. 2) and ECCS model development in accordance with the required features of the ECCS evaluation models provided in 20 CFR 50, Appendix K (Ref. 5). Therefore, this LCO provides conservative limits for other accidents.

F_Q(Z) satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO The F_Q(Z) shall be maintained within the limits of the relationships provided in the COLR.

The F_Q(Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA (Refs. 6 and 7).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F_Q(Z) limits. If F_Q(Z) cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for F_Q(Z) may produce unacceptable consequences if a design basis event occurs while F_Q(Z) is outside its specified limits.

APPLICABILITY The F_Q(Z) limits must be maintained while in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ for each 1% by which F_Q(Z) exceeds its limit maintains an acceptable absolute power density. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)



BASES

ACTIONS
(continued)

A.2

When core peaking factors are sufficiently high that LCO 3.2.1 does not permit operation at RTP, the acceptable operation limits for AFD are reduced. The acceptable operation limits are reduced 1% for each 1% by which F_Q(Z) exceeds its limit. For example, if the measured F_Q(Z) exceeds the limit by 3% and the acceptable operation limits for AFD are ± 11% at 90% RTP and ± 31% at 50% RTP, then the revised AFD Acceptable Operation Limits would be ± 8% at 90% RTP and ± 28% at 50% RTP. This ensures a near constant maximum linear heat rate in units of kilowatts per foot at the acceptable operation limits. The Completion Time of 8 hours for the change in setpoints is sufficient, considering the small likelihood of a severe transient in this relatively short time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq 1\%$ for each 1% by which F_Q(Z) exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since this trip setpoint helps protect reactor core safety limits. This reduction shall be made as follows, given an F_Q(Z) limit of 2.32, a measured F_Q(Z) of 2.4, and a Power Range Neutron Flux-High setpoint of 108%, the Power Range Neutron Flux-High setpoint must be reduced by at least 3.4% to 104.6%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

ACTIONS
(continued)

A.4

Reduction in the Overpower ΔT and Overtemperature ΔT trip setpoints by $\geq 1\%$ for each 1% by which $F_0(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since these trip setpoints help protect reactor core safety limits. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.5

Verification that $F_0(Z)$ has been restored to within its limit by performing SR 3.2.1.1 or SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.5 cannot be met within their associated Completion Times, the plant must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

Verification that F_Q(Z) is within its limit involves increasing the measured values of F_Q(Z) to allow for manufacturing tolerance and measurement uncertainties and then making a comparison with the limits. These limits are provided in the COLR. Specifically, the measured value of the Heat Flux Hot Channel Factor (F_Q^M) is increased by 3% to account for fuel manufacturing tolerances and by 5% for flux map measurement uncertainty for a full core flux map using the movable incore detector flux mapping system. This procedure is equivalent to increasing the directly measured values of F_Q(Z) by 1.0815% before comparing with LCO limits.

Performing the Surveillance in MODE 1 prior to THERMAL POWER exceeding 75% RTP after each refueling ensures that F_Q(Z) is within limit when RTP is achieved and provides confirmation of the nuclear design and the fuel loading pattern.

The Frequency of 31 EFPD is adequate for monitoring the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. Accordingly, this Frequency is short enough that the F_Q(Z) limit cannot be exceeded for any significant period of time.

SR 3.2.1.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.1.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that F₀ remains within limits and the core power distribution is consistent with the safety analyses. A Frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is $\geq 75\%$ RTP.

REFERENCES

1. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 2. 10 CFR 50.46.
 3. UFSAR, Section 15.4.5.1.
 4. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.
 5. 10 CFR 50, Appendix K.
 6. UFSAR, Section 15.6.4.1.
 7. UFSAR, Section 15.6.4.2.
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B 3.2 POWER DISTRIBUTION LIMITS

 B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

 BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location in the core during either normal operation or a postulated accident analyzed in the safety analyses.

$F_{\Delta H}^N$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F_{\Delta H}^N$ is a measure of the maximum total power produced in a fuel rod. The $F_{\Delta H}^N$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for departure from nucleate boiling (DNB).

$F_{\Delta H}^N$ is sensitive to fuel loading patterns, control bank insertion, and fuel burnup. $F_{\Delta H}^N$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F_{\Delta H}^N$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F_{\Delta H}^N$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis. $F_{\Delta H}^N$ and the QPTR LCO limit the radial component of the peaking factors.

 (continued)

BASES

BACKGROUND
(continued)

The COLR provides peaking factor limits that ensure that the design basis value for departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. All DNB limited transient events are assumed to begin with an $F_{\Delta H}^N$ value that satisfies the LCO requirements.

The design method employed to meet the DNB design criterion for fuel assemblies is the Improved Thermal Design Procedure (ITDP). With the ITDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, ITDP design limit DNBR values are determined in order to meet the DNB design criterion.

The ITDP design limit DNBR values are 1.34 and 1.33 for the typical and thimble cells, respectively, for fuel analyses with the WRB-2 correlation.

DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR values are 1.52 and 1.51 for the typical and thimble cells, respectively.

(continued)

BASES

 BACKGROUND
 (continued)

For both the WRB-1 and WRB-2 correlations, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation applies in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

 APPLICABLE
 SAFETY ANALYSES

Limits on $F_{\Delta H}^N$ preclude core power distributions that exceed the following fuel design limits:

- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);
- c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

 (continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}^N$ are the core parameters of most importance. The limits on $F_{\Delta H}^N$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency (i.e., Condition 1 events as described in Reference 4). The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion.

The allowable $F_{\Delta H}^N$ limit increases with decreasing power level. This functionality in $F_{\Delta H}^N$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}^N$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F_{\Delta H}^N$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F_{\Delta H}^N$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F_{\Delta H}^N$ is measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR; and Bank Insertion, Sequence and Overlap Limits.

$F_{\Delta H}^N$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

$F_{\Delta H}^N$ shall be maintained within the limits of the relationship provided in the COLR.

The $F_{\Delta H}^N$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB.

The limiting value of $F_{\Delta H}^N$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F_{\Delta H}^N$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F_{\Delta H}^N$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F_{\Delta H}^N$ in MODES 2, 3, 4, and 5 have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}^N$ in these modes.

(continued)

BASES (continued)

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its limit maintains an acceptable DNBR margin. When the $F_{\Delta H}^N$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F_{\Delta H}^N$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Reducing THERMAL POWER increases the DNB margin. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions and ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. This reduction shall be made as follows, given that the $F_{\Delta H}^N$ limit is exceeded by 3% and the Power Range Neutron Flux-High setpoint is 108%, the Power Range Neutron Flux-High setpoint must be reduced by at least 3% to 105%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with required action A.1.

A.3

Reduction in the Overpower ΔT and Overttemperature ΔT trip setpoints by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its limit, ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

 (continued)



BASES

ACTIONS
(continued)

A.4

Verification that $F_{\Delta H}^N$ has been restored within its limit by performing SR 3.2.2.1 or SR 3.2.2.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that the cause that led to the $F_{\Delta H}^N$ exceeding its limit is corrected, and core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.4 cannot be met within their associated Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

The allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1

The value of $F_{\Delta H}^N$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F_{\Delta H}^N$ from the measured flux distributions. The measured value of $F_{\Delta H}^N$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F_{\Delta H}^N$ limit.

After each refueling, $F_{\Delta H}^N$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F_{\Delta H}^N$ limits are met at the beginning of each fuel cycle.

(continued)

BASES

 SURVEILLANCE
 REQUIREMENTS

SR 3.2.2.1 (continued)

The Frequency of 31 EFPD is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}^N$ limit cannot be exceeded for any significant period of operation. When the plant is already performing SR 3.2.2.2 to satisfy other requirements, SR 3.2.2.2 does not need to be suspended in order to perform SR 3.2.2.1 since the performance of SR 3.2.2.2 meets the requirements of SR 3.2.2.1.

SR 3.2.2.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.2.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that $F_{\Delta H}^N$ remains within limits and the core power distribution is consistent with the safety analyses. A Frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is $\geq 75\%$ RTP.

 (continued)



BASES (continued)

- REFERENCES
1. 10 CFR 50.46.
 2. UFSAR, Section 15.4.5.1.
 3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10 1967.
 4. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during plant maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) and QUADRANT POWER TILT RATIO (QPTR) LCOs limit the radial component of the peaking factors.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Ref. 1) entails:

- a. Establishing an envelope of allowed power shapes and power densities;
- b. Devising an operating strategy for the cycle that maximizes plant flexibility (maneuvering) and minimizes axial power shape changes;
- c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and
- d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events (Ref. 2). This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overttemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator to help define the power profile from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom excore neutron detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\% \Delta$ flux or $\% \Delta I$.

With THERMAL POWER $\geq 90\%$ RTP (i.e., Part A of this LCO), the AFD must be kept within the target band about the target flux difference. The target band is provided in the COLR. With the AFD outside the target band with THERMAL POWER $\geq 90\%$ RTP, the assumptions of the accident analyses may be violated. With THERMAL POWER $< 90\%$ RTP, the AFD may be outside the target band provided that the deviation time is restricted.

It is intended that the plant is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is $\geq 50\%$ RTP and $< 90\%$ RTP (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours when $> 15\%$ RTP, is allowed during which the plant may be operated outside of the target band but within the acceptable operation limits provided in the COLR. The cumulative penalty time is the sum of penalty times as calculated by Notes 2 and 3 of this LCO.

(continued)

BASES

LCO
(continued)

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The reduced penalty deviation time accumulation rate reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the Plant Process Computer System (PPCS) is nominally once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The inoperability of this monitor requires independent verification that AFD remains within limit and that the peaking factors assumed in the accident analyses remain valid.

(continued)

BASES

LCO
(continued)

This LCO is modified by four Notes. The first Note states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. The average of the four OPERABLE excore detectors is used to determine when AFD is outside the target band. If one excore detector is out of service, the remaining three detectors are used to derive the average AFD. The second and third Notes describe how the cumulative penalty deviation time is calculated. The second Note states that with THERMAL POWER \geq 50% RTP the penalty deviation time is accumulated at the rate of 1 minute for each 1 minute of power operation with AFD outside the target band. The third Note states that with THERMAL POWER $>$ 15% RTP and $<$ 50% RTP the penalty deviation time is accumulated at the rate of 0.5 minutes for each 1 minute of power operation with AFD outside the target band. The cumulative penalty time is the sum of penalty times from Notes 2 and 3 of this LCO. The fourth Note addresses AFD outside of the target band during surveillances. For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

(continued)

BASES (continued)

APPLICABILITY AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1). Above 15% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. Also, low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. The value of the AFD in these conditions does not affect the consequences of the design basis events.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER $\geq 90\%$ RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If Required Action A.1 is not completed with the required Completion Time of 15 minutes, the axial xenon distribution starts to become skewed. Reducing THERMAL POWER to $< 90\%$ RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes to reduce THERMAL POWER to $< 90\%$ RTP allows for a controlled reduction in power without allowing the plant to remain in an unanalyzed condition for an extended period of time.

(continued)

BASES

ACTIONS
(continued)C.1

This Required Action must be implemented with THERMAL POWER < 90% RTP but \geq 50% RTP if either the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

With THERMAL POWER < 90% RTP but \geq 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. Reducing THERMAL POWER to < 50% RTP will put the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits. The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

D.1

When the AFD monitor alarm is inoperable and THERMAL POWER is \geq 90% RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every 15 minutes to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 15 minutes is adequate to ensure that the AFD is within its limits at high THERMAL POWER levels and is consistent with the Completion Time for restoring AFD to within limits (Condition A).

(continued)

BASES

ACTIONS
(continued)E.1

When the AFD monitor alarm is inoperable and THERMAL POWER is $< 90\%$ RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every hour to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SURVEILLANCE
REQUIREMENTSSR 3.2.3.1

This SR is the verification that the AFD monitor is OPERABLE. This is normally accomplished by introducing a signal into the plant process computer to verify control room annunciation of AFD not within the target band. The Frequency of 12 hours is sufficient to ensure OPERABILITY of the AFD monitor since under normal plant operation, the AFD is not expected to significantly change.

SR 3.2.3.2

The AFD is monitored on a continuous basis using the Plant Process Computer System (PPCS) that has an AFD monitor alarm. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control board annunciator immediately if the average AFD is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.3.2 (continued)

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at $\geq 90\%$ RTP, the AFD measurement is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. The AFD should be monitored and logged more frequently during periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.2 is modified by two Notes. The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER $\geq 90\%$ RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

SR 3.2.3.3

The AFD is monitored on a continuous basis using the PPCS that has an AFD monitor alarm. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control board annunciator immediately if the average AFD is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.

(continued)



BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.3.3 (continued)

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at < 90% RTP, but > 15% RTP, the AFD measurement is monitored at a Surveillance Frequency of 1 hour to ensure that the AFD is within its limits. The Frequency of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3 is modified by two Notes. The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER < 90% RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

SR 3.2.3.4

This Surveillance requires that the target flux difference be updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.3.4 (continued)

There are two methods by which this update can be completed. The first method requires measuring the target flux difference in accordance with SR 3.2.3.5. This measurement may be obtained using incore or excore instrumentation. The second method involves interpolation between measured and predicted values. The nuclear design report provides predicted values for target flux difference at various cycle burnups. The difference between the last measured value and the predicted value at the same burnup is applied to the predicted value at the burnup where the target flux difference update is required. This revised predicted value can then be used to determine the updated value of the target flux difference.

SR 3.2.3.5

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore-excore calibrations that may have occurred in the interim.

This SR is modified by a Note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

(continued)

BASES (continued)

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
 2. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 3. UFSAR, Section 7.7.2.6.4.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. Quadrant Power Tilt is a core tilt that is measured with the use of the excore power range flux detectors. A core tilt is defined as the ratio of maximum to average quadrant power. The QPTR is defined as the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants. Limiting the QPTR prevents radial xenon oscillations and will indicate any core asymmetries.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE
SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$), and Bank Insertion, Sequence and Overlap Limits are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F_{\Delta H}^N$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F_{\Delta H}^N$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR monitor alarm shall be OPERABLE and QPTR shall be maintained at or below the limit of 1.02.

QPTR is monitored on an automatic basis using the Plant Process Computer System (PPCS) that has a QPTR monitor alarm. The PPCS determines from the excore detector outputs the ratio of the highest average nuclear power in any quadrant to the average of nuclear power in the four quadrants and provides an alarm message if the QPTR is above the 1.02 limit.

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.025 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F_{\Delta H}^N$ is possibly challenged. However, the additional QPTR of 0.005 is provided for margin in the LCO.

(continued)

BASES (continued)

APPLICABILITY The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits assumed in the safety analyses.

Applicability in MODE 1 \leq 50% RTP and in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F_{\Delta H}^N$ and $F_Q(Z)$ LCOs still apply below 50% RTP, but allow progressively higher peaking factors as THERMAL POWER decreases below 50% RTP.

ACTIONS

A.1

With the QPTR exceeding its limit, limiting THERMAL POWER to \geq 3% below RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. A further increase in the QPTR requires a lower limit to THERMAL POWER in accordance with Required Action A.2.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR in accordance with SR 3.2.4.1 once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER must be limited accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

(continued)

BASES

ACTIONS
(continued)

A.3

The peaking factors $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing Srs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

(continued)

BASES

ACTIONS
(continued)

A.5

If the QPTR has exceeded the 1.02 limit and the verification of $F_{\Delta H}^N$ and $F_Q(Z)$ shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to detect any subsequent significant changes in QPTR and to provide a meaningful QPTR alarm.

Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). It is necessary to verify that the core power distribution is acceptable prior to adjusting the excore detectors to eliminate the indicated tilt and increasing power to ensure that the plant is not operating in an unanalyzed condition. This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

After the flux tilt is normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F_{\Delta H}^N$ are within their specified limits within 24 hours after reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but it increases slowly, then the peaking factor surveillances must be performed within 24 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

(continued)

BASES

ACTIONS

A.6 (continued)

Required Action A.5 is modified by three Notes. The first Note states that it is not necessary to perform Required Action A.5 if the cause of the QPTR alarm is associated with instrumentation alignment. The intent of this Note is to clarify that the core power distribution does not have to be re-verified if the QPTR alarm is only due to the instrumentation (i.e., the excore detectors) being out of adjustment and not due to an anomaly within the core. The second Note states that the peaking factor surveillances are not required until after the excore detectors have been normalized to eliminate the indicated tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are adjusted to eliminate the indicated tilt and the core returned to power. The third Note states that only one of the following Completion Times, whichever becomes applicable first, must be met. The intent of this Note is to clearly indicate that the first Completion Time to become applicable is the Completion Time which must be met to satisfy Required Action A.6.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the plant must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

When the QPTR monitor alarm is inoperable the QPTR must be verified within limits at a frequency of every 24 hours to ensure that the plant does not operate in an unanalyzed condition. When THERMAL POWER is $\geq 75\%$ RTP and one power range channel is inoperable, QPTR cannot be adequately measured using the excore detectors. In this situation a flux map must be completed to verify that the core power distribution is consistent with the safety analyses. A Completion Time of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt and provides sufficient time to stabilize the plant and perform a flux map when necessary. The Completion Time of 24 hours is also consistent with the Frequency of SR 3.2.4.3 with one inoperable power range channel since these channels provide input into the QPTR monitor.

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1

This SR is the verification that the QPTR monitor is OPERABLE. This is normally accomplished by introducing a signal into the PPCS to verify control room annunciation of QPTR not within limit. The Frequency of 12 hours is sufficient to ensure OPERABILITY of the QPTR monitor since under normal plant operation, QPTR is not expected to significantly change.

SR 3.2.4.2

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.2 (continued)

SR 3.2.4.2 is modified by two Notes. The first allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. The second Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is \geq 75% RTP and one power range channel is inoperable. The intent of this Note is to clarify that when one power range channel is inoperable and THERMAL POWER is \geq 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. At or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_{\alpha}(Z)$ and $F_{\Delta H}^N$ remain within limits and the core power distribution is consistent with the safety analyses.

SR 3.2.4.3

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits when the QPTR alarm is inoperable. The Frequency of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.3 (continued)

SR 3.2.4.3 is modified by three Notes. The first Note states that the surveillance is only required to be performed if the QPTR monitor alarm is inoperable. This surveillance requires a more frequent verification that the QPTR is within limit since the monitor alarm is inoperable. The second Note allows QPTR to be calculated with three power range channels if THERMAL POWER is $< 75\%$ RTP and one power range channel is inoperable. The third Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is $\geq 75\%$ RTP and one power range channel is inoperable. The intent of this Note is clarify that when one power range channel is inoperable and THERMAL POWER is $\geq 75\%$ RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. At or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_Q(Z)$ and $F_{\Delta H}^N$ remain within limits and the core power distribution is consistent with the safety analyses.

REFERENCES

1. 10 CFR 50.46.
 2. UFSAR, Section 15.4.5.
 3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.
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3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable. <u>OR</u> Two source range channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. As required by Required Action A.1 and referenced by Table 3.3.1-1.	B.1 Restore channel to OPERABLE status.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required Action and associated Completion Time of Condition B not met.</p>	<p>C.1 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u></p> <p>C.2 Initiate action to fully insert all rods.</p>	<p>6 hours</p>
	<p><u>AND</u></p> <p>C.3 Place Control Rod Drive System in a condition incapable of rod withdrawal.</p>	<p>7 hours</p>
<p>D. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>D.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>H. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>H.1 Restore at least one channel to OPERABLE status upon discovery of two inoperable channels.</p>	<p>1 hour from discovery of two inoperable channels</p>
	<p><u>AND</u></p>	
	<p>H.2 Suspend operations involving positive reactivity additions.</p>	<p>Immediately</p>
<p>I. Required Action and associated Completion Time of Condition H not met.</p>	<p><u>AND</u></p>	
	<p>H.3 Restore channel to OPERABLE status.</p>	<p>48 hours</p>
	<p>I.1 Initiate action to fully insert all rods.</p>	<p>Immediately</p>
	<p>I.2 Place the Control Rod Drive System in a condition incapable of rod withdrawal.</p>	<p>1 hour</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>J. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>J.1 Suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>J.2 Perform SR 3.1.1.1.</p>	<p>Immediately</p> <p>12 hours</p> <p><u>AND</u></p> <p>Once per 12 hours thereafter</p>
<p>K. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>K.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>Place channel in trip.</p>	<p>6 hours</p>
<p>L. Required Action and associated Completion Time of Condition K not met.</p>	<p>L.1 Reduce THERMAL POWER to < 8.5% RTP.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>M. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>M.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.</p>	<p>6 hours</p>
<p>N. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>N.1 Restore channel to OPERABLE status.</p>	<p>6 hours</p>
<p>O. Required Action and associated Completion Time of Condition M or N not met.</p>	<p>O.1 Reduce THERMAL POWER to < 50% RTP.</p>	<p>6 hours</p>
<p>P. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>P.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>Q. Required Action and Associated Completion Time of Condition P not met.</p>	<p>Q.1 Reduce THERMAL POWER to < 50% RTP.</p> <p><u>AND</u></p> <p>Q.2.1 Verify Steam Dump System is OPERABLE.</p> <p><u>OR</u></p> <p>Q.2.2 Reduce THERMAL POWER to < 8% RTP.</p>	<p>6 hours</p> <p>7 hours</p> <p>7 hours</p>
<p>R. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>R.1 -----NOTE----- The inoperable train may be bypassed for up to 4 hours for surveillance testing of the other train. -----</p> <p>Restore train to OPERABLE status.</p>	<p>6 hours</p>
<p>S. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>S.1 Verify interlock is in required state for existing plant conditions.</p> <p><u>OR</u></p> <p>S.2 Declare associated RTS Function channel(s) inoperable.</p>	<p>1 hour</p> <p>1 hour</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>T. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>-----NOTES----- 1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. 2. One RTB may be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE. -----</p> <p>T.1 Restore train to OPERABLE status.</p>	<p>1 hour</p>
<p>U. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>U.1 Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.</p> <p><u>AND</u></p> <p>U.2 Restore trip mechanism to OPERABLE status.</p>	<p>1 hour from discovery of two inoperable trip mechanisms</p> <p>48 hours</p>
<p>V. Required Action and associated Completion Time of Condition R, S, T, or U not met.</p>	<p>V.1 Be in MODE 3.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>W. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>W.1 Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.</p>	<p>1 hour from discovery of two inoperable trip mechanisms</p>
	<p><u>AND</u></p> <p>W.2 Restore trip mechanism or train to OPERABLE status.</p>	<p>48 hours</p>
<p>X. Required Action and associated Completion Time of Condition W not met.</p>	<p>X.1 Initiate action to fully insert all rods.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>X.2 Place the Control Rod Drive System in a Condition incapable of rod withdrawal.</p>	<p>1 hour</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-----NOTE----- Required to be performed within 12 hours after THERMAL POWER is \geq 50% RTP. Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output and adjust if calorimetric power is $>$ 2% higher than indicated NIS power.	24 hours
SR 3.3.1.3	-----NOTES----- 1. Required to be performed within 7 days after THERMAL POWER is \geq 50% RTP but prior to exceeding 90% RTP following each refueling and if the Surveillance has not been performed within the last 31 EFPD. 2. Performance of SR 3.3.1.6 satisfies this SR. ----- Compare results of the incore detector measurements to NIS AFD and adjust if absolute difference is \geq 3%.	31 effective full power days (EFPD)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.1.4 Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.6 -----NOTE----- Not required to be performed until 7 days after THERMAL POWER is \geq 50% RTP, but prior to exceeding 90% RTP following each refueling. ----- Calibrate excore channels to agree with incore detector measurements.	92 EFPD
SR 3.3.1.7 -----NOTE----- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3. ----- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.8 -----NOTE-----</p> <ol style="list-style-type: none"> 1. Not required for power range and intermediate range instrumentation until 4 hours after reducing power < 6% RTP. 2. Not required for source range instrumentation until 4 hours after reducing power < 5E-11 amps. <p>-----</p> <p>Perform COT.</p>	<p>92 days</p>
<p>SR 3.3.1.9 -----NOTE-----</p> <p>Setpoint verification is not required.</p> <p>-----</p> <p>Perform TADOT.</p>	<p>92 days</p>
<p>SR 3.3.1.10 -----NOTE-----</p> <p>Neutron detectors are excluded.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	<p>24 months</p>
<p>SR 3.3.1.11 Perform TADOT.</p>	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.1.12 -----NOTE----- Setpoint verification is not required. ----- Perform TADOT.	Prior to reactor startup if not performed within previous 31 days
SR 3.3.1.13 Perform COT.	24 months

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Reactor Trip	3(a), 4 ^{1,2} (a), 5(a)	2	B,C	SR 3.3.1.11	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D,G	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10	≤ 109% RTP
b. Low	1(b),2	4	D,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 25% RTP
3. Intermediate Range Neutron Flux	1(b), 2	2	E,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
4. Source Range Neutron Flux	2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
	3(a), 4(a), 5(a)	2	H,I	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	(d)
	3(e), 4(e), 5(e)	1	J	SR 3.3.1.1 SR 3.3.1.10	NA

(continued)

- (a) With Control Rod Drive (CRD) System capable of rod withdrawal, or all rods not fully inserted.
- (b) THERMAL POWER < 6% RTP.
- (c) Both Intermediate Range channels < 5E-11 amps.
- (d) UFSAR Table 7.2-3.
- (e) With CRD System incapable of withdrawal and all rods fully inserted. In this condition, the Source Range Neutron Flux function does not provide a reactor trip, only indication.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
5. Overtemperature ΔT	1,2	4	D,G	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 1 (page 3.3-18)
6. Overpower ΔT	1,2	4	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 2 (page 3.3-19)
7. Pressurizer Pressure					
a. Low	1(f)	4	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1865 psig
b. High	1,2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2385 psig
8. Pressurizer Water Level -High	1,2	3	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\leq 88\%$
9. Reactor Coolant Flow -Low					
a. Single Loop	1(g)	3 per loop	H,O	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\geq 90\%$
b. Two Loops	1(h)	3 per loop	K,L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\geq 90\%$

(continued)

(f) THERMAL POWER $\geq 8.5\%$ RTP.

(g) THERMAL POWER $\geq 50\%$ RTP.

(h) THERMAL POWER $\geq 8.5\%$ RTP and Reactor Coolant Flow -Low (Single Loop) trip Function blocked.

Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	1(g)	1 per RCP	H,O	SR 3.3.1.11	NA
b. Two Loops	1(i)	1 per RCP	K,L	SR 3.3.1.11	NA
11. Undervoltage – Bus 11A and 11B	1(f)	2 per bus	K,L	SR 3.3.1.9 SR 3.3.1.10	(d)
12. Underfrequency Bus 11A and 11B	1(f)	2 per bus	K,L	SR 3.3.1.9 SR 3.3.1.10	≥ 57.5 HZ
13. Steam Generator (SG) Water Level –Low Low	1,2	3 per SG	D,G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 16%
14. Turbine Trip					
a. Low Autostop Oil Pressure	1(j)(k)	3	P,Q	SR 3.3.1.10 SR 3.3.1.12	(d)
b. Turbine Stop Valve Closure	1(j)(k)	2	P,Q	SR 3.3.1.12	NA
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2	R,V	SR 3.3.1.11	NA

(continued)

- (d) UFSAR Table 7.2-3.
- (f) THERMAL POWER ≥ 8.5% RTP.
- (g) THERMAL POWER ≥ 50% RTP.
- (i) THERMAL POWER ≥ 8.5% RTP and RCP Breaker Position (Single Loop) trip Function blocked.
- (j) THERMAL POWER > 8% RTP, and either no circulating water pump breakers closed, or condenser vacuum ≤ 20".
- (k) THERMAL POWER ≥ 50% RTP, 1 of 2 circulating water pump breakers closed, and condenser vacuum > 20".

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
16. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(c)	2	S,V	SR 3.3.1.10 SR 3.3.1.13	$\geq 5E-11$ amp
b. Low Power Reactor Trips Block, P-7	1(f)	4 (power range only)	S,V	SR 3.3.1.10 SR 3.3.1.13	< 8.5% RTP
c. Power Range Neutron Flux, P-8	1(g)	4	S,V	SR 3.3.1.10 SR 3.3.1.13	< 50% RTP
d. Power Range Neutron Flux, P-9	1(k)	4	S,V	SR 3.3.1.10 SR 3.3.1.13	< 50% RTP
	1(j)	4	S,V	SR 3.3.1.10 SR 3.3.1.13	$\leq 8\%$ RTP
e. Power Range Neutron Flux, P-10	1(b), 2	4	S,V	SR 3.3.1.10 SR 3.3.1.13	$\geq 6\%$ RTP
17. Reactor Trip Breakers	1, 2	2 trains	T,V	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	W,X	SR 3.3.1.4	NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2	1 each per RTB	U,V	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	W,X	SR 3.3.1.4	NA
19. Automatic Trip Logic	1, 2	2 trains	R,V	SR 3.3.1.5	NA
	3(a), 4(a), 5(a)	2 trains	W,X	SR 3.3.1.5	NA

(a) With CRD System capable of rod withdrawal or all rods not fully inserted.

(b) THERMAL POWER < 6% RTP.

(c) Both Intermediate Range channels < 5E-11 amps.

(f) THERMAL POWER $\geq 8.5\%$ RTP.

(g) THERMAL POWER $\geq 50\%$ RTP.

(j) THERMAL POWER > 8% RTP, and either no circulating water pump breakers closed, or condenser vacuum $\leq 20"$.

(k) THERMAL POWER $\geq 50\%$ RTP, 1 of 2 circulating water pump breakers closed, and condenser vacuum > 20".

(l) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Trip Setpoint is defined by:

$$\text{Overtemperature } \Delta T \leq \Delta T_o \left\{ K_1 + K_2 (P - P') - K_3 (T - T') \left[\frac{1 + \tau_1 s}{1 + \tau_2 s} \right] - f(\Delta I) \right\}$$

Where:

ΔT is measured RCS ΔT , °F.

ΔT_o is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1} .

T is the measured RCS average temperature, °F.

T' is the nominal T_{avg} at RTP, °F.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure, psig.

K_1 is the Overtemperature ΔT reactor trip setpoint, 1.20.

K_2 is the Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, 0.000900.

K_3 is the Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, 0.0209.

τ_1 is the measured lead/lag time constant, 25 seconds.

τ_2 is the measured lead/lag time constant, 5 seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$$f(\Delta I) = 0 \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

$$f(\Delta I) = 1.3 \{ (q_t - q_b) - 13 \} \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Trip Setpoint is defined by:

$$\text{Overpower } \Delta T \leq \Delta T_o \left\{ K_4 - K_5 (T - T') - K_6 \left[\frac{\tau_3 s T}{\tau_3 s + 1} \right] - f(\Delta I) \right\}$$

Where:

ΔT is measured RCS ΔT , °F.

ΔT_o is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1} .

T is the measured RCS average temperature, °F.

T' is the nominal T_{avg} at RTP, °F.

K_4 is the Overpower ΔT reactor trip setpoint, 1.077.

K_5 is the Overpower ΔT reactor trip heatup setpoint penalty coefficient which is:

0.0 for $T < T^2$ and;

0.0011 for $T \geq T^2$.

K_6 is the Overpower ΔT reactor trip thermal time delay setpoint penalty which is:

0.0262 for increasing T and;

0.00 for decreasing T .

τ_3 is the measured lead/lag time constant, 10 seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$$f(\Delta I) = 0 \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

$$f(\Delta I) = 1.3 \{ (q_t - q_b) - 13 \} \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel or train.	Immediately
B. As required by Required Action A.1 and referenced by Table 3.3.2-1.	B.1 Restore channel to OPERABLE status.	48 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced by Table 3.3.2-1.	D.1 Restore channel to OPERABLE status.	48 hours
E. As required by Required Action A.1 and referenced by Table 3.3.2-1.	E.1 Restore train to OPERABLE status.	6 hours
F. As required by Required Action A.1 and referenced by Table 3.3.2-1.	F.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels. ----- Place channel in trip.	6 hours
G. Required Action and associated Completion Time of Condition D, E, or F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	6 hours 12 hours
H. As required by Required Action A.1 and referenced by Table 3.3.2-1.	H.1 Restore channel to OPERABLE status.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action A.1 and referenced by Table 3.3.2-1.	I.1 Restore train to OPERABLE status.	6 hours
J. As required by Required Action A.1 and referenced by Table 3.3.2-1.	J.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels. ----- Place channel in trip.	6 hours
K. Required Action and associated Completion Time of Condition H, I, or J not met.	K.1 Be in MODE 3. <u>AND</u> K.2 Be in MODE 5.	6 hours 36 hours
L. As required by Required Action A.1 and referenced by Table 3.3.2-1.	L.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels. ----- Place channel in trip.	6 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>M. Required Action and associated Completion Time of Condition L not met.</p>	<p>M.1 Be in MODE 3. <u>AND</u> M.2 Reduce pressurizer pressure to < 2000 psig.</p>	<p>6 hours 12 hours</p>
<p>N. As required by Required Action A.1 and referenced by Table 3.3.2-1.</p>	<p>N.1 Declare associated Auxiliary Feedwater pump inoperable and enter applicable condition(s) of LCO 3.7.5, "Auxiliary Feedwater (AFW) System."</p>	<p>Immediately</p>



SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.
-

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform COT.	92 days
SR 3.3.2.3	-----NOTE----- Verification of relay setpoints not required. ----- Perform TADOT.	92 days
SR 3.3.2.4	-----NOTE----- Verification of relay setpoints not required. ----- Perform TADOT.	24 months
SR 3.3.2.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.6	Verify the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig.	24 months
SR 3.3.2.7	Perform ACTUATION LOGIC TEST.	24 months

Table 3.3.2-1 (page 1 of 3)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3	2	D,G	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	I,K	SR 3.3.2.7	NA	NA
c. Containment Pressure -High	1,2,3,4	3	J,K	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 6.0 psig	≤ 4.0 psig
d. Pressurizer Pressure -Low	1,2,3 ^(a)	3	L,M	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5 SR 3.3.2.6	≥ 1715 psig	≥ 1750 psig
e. Steam Line Pressure -Low	1,2,3 ^(a)	3 per steam line	L,M	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5 SR 3.3.2.6	≥ 358 psig	≥ 514 psig
2. Containment Spray						
a. Manual Initiation						
Left pushbutton	1,2,3,4	1	H,K	SR 3.3.2.4	NA	NA
Right pushbutton	1,2,3,4	1	H,K	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	I,K	SR 3.3.2.7	NA	NA
c. Containment Pressure -High High	1,2,3,4	3 per set	J,K	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 32.5 psig	≤ 28 psig
3. Containment Isolation						
a. Manual Initiation	1,2,3,4	2	H,K	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	I,K	SR 3.3.2.7	NA	NA
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(a) Pressurizer Pressure ≥ 2000 psig.

Table 3.3.2-1 (page 2 of 3)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation						
a. Manual Initiation	1,2 ^(b) ,3 ^(b)	1 per loop	D,G	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2 ^(b) ,3 ^(b)	2 trains	E,G	SR 3.3.2.7	NA	NA
c. Containment Pressure -High High	1,2 ^(b) ,3 ^(b)	3	F,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 20 psig	≤ 18 psig
d. High Steam Flow	1,2 ^(b) ,3 ^(b)	2 per steam line	F,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 0.55E6 lbm/hr @ 755 psi	≤ 0.4E6 lbm/hr @ 755 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with T _{avg} -Low	1,2 ^(b) ,3 ^(b)	2 per loop	F,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≥ 543°F	≥ 545°F
e. High -High Steam Flow	1,2 ^(b) ,3 ^(b)	2 per steam line	F,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 3.7E6 lbm/hr @ 755 psig	≤ 3.6E6 lbm/hr @ 755 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(b) Except when both MSIVs are closed and de-activated.

Table 3.3.2-1 (page 3 of 3)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2 ^(c) ,3 ^(c)	2 trains	E,G	SR 3.3.2.7	NA	NA
b. SG Water Level -High	1,2 ^(c) ,3 ^(c)	3 per SG	F,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 68%	≤ 67%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater (AFW)						
a. Manual Initiation						
AFW	1,2,3	1 per pump	N	SR 3.3.2.4	NA	NA
Standby AFW	1,2,3	1 per pump	N	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	E,G	SR 3.3.2.7	NA	NA
c. SG Water Level -Low Low	1,2,3	3 per SG	D,G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≥ 16%	≥ 17%
d. Safety Injection (Motor driven pumps only)	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
e. Undervoltage -Bus 11A and 11B (Turbine driven pump only)	1,2,3	2 per bus	D,G	SR 3.3.2.3 SR 3.3.2.5	≥ 2450 V with ≤ 3.6 sec time delay	≥ 2579 V with ≤ 3.6 sec time delay
f. Trip of Both Main Feedwater Pumps (Motor driven pumps only)	1,2	2 per MFW pump	B,C	SR 3.3.2.4	NA	NA

(c) Except when all Main Feedwater Regulating and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Not applicable to Functions 3 and 4. -----</p> <p>One or more Functions with one required channel inoperable..</p>	<p>A.1 Restore required channel to OPERABLE status.</p>	<p>30 days</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Initiate action to prepare and submit a special report.</p>	<p>Immediately</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Only applicable to Functions 3 and 4. ----- One or more Functions with required channel inoperable.</p>	<p>C.1 Restore required channel to OPERABLE status.</p>	<p>7 days</p>
<p>D. -----NOTE----- Not applicable to Function 11. ----- One or more Functions with two required channels inoperable.</p>	<p>D.1 Restore one channel to OPERABLE status.</p>	<p>7 days</p>
<p>E. Two hydrogen monitor channels inoperable.</p>	<p>E.1 Restore one hydrogen monitor channel to OPERABLE status.</p>	<p>72 hours</p>
<p>F. Required Action and associated Completion Time of Condition C, D, or E not met.</p>	<p>F.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.</p>	<p>Immediately</p>
<p>G. As required by Required Action F.1 and referenced in Table 3.3.3-1.</p>	<p>G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.</p>	<p>6 hours 12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action F.1 and referenced in Table 3.3.3-1.	H.1 Initiate action to prepare and submit a special report.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in
Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	Perform CHANNEL CALIBRATION.	24 months

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION
1. Pressurizer Pressure	2	G
2. Pressurizer Level	2	G
3. Reactor Coolant System (RCS) Hot Leg Temperature	1 per loop	G
4. RCS Cold Leg Temperature	1 per loop	G
5. RCS Pressure (Wide Range)	2	G
6. RCS Subcooling Monitor	2	G
7. Reactor Vessel Water Level	2	H
8. Containment Sump B Water Level	2	G
9. Containment Pressure (Wide Range)	2	G
10. Containment Area Radiation (High Range)	2	H
11. Hydrogen Monitors	2	G
12. Condensate Storage Tank Level	2	G
13. Refueling Water Storage Tank Level	2	G
14. Residual Heat Removal Flow	2	G
15. Core Exit Temperature—Quadrant 1	2(a)	G
16. Core Exit Temperature—Quadrant 2	2(a)	G
17. Core Exit Temperature—Quadrant 3	2(a)	G
18. Core Exit Temperature—Quadrant 4	2(a)	G
19. Auxiliary Feedwater (AFW) Flow to Steam Generator (SG) A	2	G
20. AFW Flow to SG B	2	G
21. SG Water Level (Narrow Range) to SG A	2	G
22. SG Water Level (Narrow Range) to SG B	2	G

(continued)

(a) A channel consists of two core exit thermocouples (CETs).

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION
23. SG Water Level (Wide Range) to SG A	2	G
24. SG Water Level (Wide Range) to SG B	2	G
25. SG Pressure to SG A	2	G
26. SG Pressure to SG B	2	G

3.3 INSTRUMENTATION

3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.4 Each 480 V safeguards bus shall have two OPERABLE channels of LOP DG Start Instrumentation.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2,
"AC Sources - MODES 5 and 6."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each 480 V safeguards bus.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more 480 V bus(es) with one channel inoperable.	A.1 Place channel(s) in trip.	6 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> One or more 480 V bus(es) with two channels inoperable.	B.1 Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 4 hours provided the second channel maintains LOP DG start capability.

SURVEILLANCE		FREQUENCY																		
SR 3.3.4.1	Perform TADOT.	31 days																		
SR 3.3.4.2	Perform CHANNEL CALIBRATION with Trip Setpoint and Allowable Value for each 480 V bus as follows: a. Loss of voltage: <table style="margin-left: 40px;"> <thead> <tr> <th></th> <th style="text-align: center;"><u>Allowable Value</u></th> <th style="text-align: center;"><u>Trip Setpoint</u></th> </tr> </thead> <tbody> <tr> <td>Bus voltage</td> <td>$\geq 368 \text{ V}$</td> <td>$\geq 372.8 \text{ V}$</td> </tr> <tr> <td>Time delay</td> <td>$\leq 2.75 \text{ sec}$</td> <td>$2.4 \pm 0.12 \text{ sec}$</td> </tr> </tbody> </table> b. Degraded voltage: <table style="margin-left: 40px;"> <thead> <tr> <th></th> <th style="text-align: center;"><u>Allowable Value</u></th> <th style="text-align: center;"><u>Trip Setpoint</u></th> </tr> </thead> <tbody> <tr> <td>Bus voltage</td> <td>$\geq 414 \text{ V}$</td> <td>$\geq 419.2 \text{ V}$</td> </tr> <tr> <td>Time delay</td> <td>$\leq 1520 \text{ sec}$</td> <td>$\leq 1520 \text{ sec}$</td> </tr> </tbody> </table>		<u>Allowable Value</u>	<u>Trip Setpoint</u>	Bus voltage	$\geq 368 \text{ V}$	$\geq 372.8 \text{ V}$	Time delay	$\leq 2.75 \text{ sec}$	$2.4 \pm 0.12 \text{ sec}$		<u>Allowable Value</u>	<u>Trip Setpoint</u>	Bus voltage	$\geq 414 \text{ V}$	$\geq 419.2 \text{ V}$	Time delay	$\leq 1520 \text{ sec}$	$\leq 1520 \text{ sec}$	24 months
	<u>Allowable Value</u>	<u>Trip Setpoint</u>																		
Bus voltage	$\geq 368 \text{ V}$	$\geq 372.8 \text{ V}$																		
Time delay	$\leq 2.75 \text{ sec}$	$2.4 \pm 0.12 \text{ sec}$																		
	<u>Allowable Value</u>	<u>Trip Setpoint</u>																		
Bus voltage	$\geq 414 \text{ V}$	$\geq 419.2 \text{ V}$																		
Time delay	$\leq 1520 \text{ sec}$	$\leq 1520 \text{ sec}$																		

3.3 INSTRUMENTATION

3.3.5 Containment Ventilation Isolation Instrumentation

LC0 3.3.5 The Containment Ventilation Isolation instrumentation for each Function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
 During CORE ALTERATIONS,
 During movement of irradiated fuel assemblies within
 containment.

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. -----NOTE----- Only applicable in MODE 1, 2, 3, or 4. -----</p> <p>One or more Functions with one or more manual or automatic actuation trains inoperable.</p> <p><u>OR</u></p> <p>Both radiation monitoring channels inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1</p> <p>Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Boundaries," for containment mini- purge isolation valves made inoperable by isolation instrumentation.</p>	<p>Immediately</p>

(continued)

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.5-1 to determine which SRs apply for each Containment
Ventilation Isolation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.5.1 Perform CHANNEL CHECK.	24 hours
SR 3.3.5.2 Perform COT.	92 days
SR 3.3.5.3 Perform ACTUATION LOGIC TEST.	24 months
SR 3.3.5.4 Perform CHANNEL CALIBRATION.	24 months

Containment Ventilation Isolation Instrumentation
3.3.5

Table 3.3.5-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.5.3	NA
2. Containment Radiation			
a. Gaseous	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
b. Particulate	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
3. Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3, for all initiation functions and requirements.		
4. Containment Spray -Manual Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 2.a, for all initiation functions and requirements.		

Notes:

(a) Per Radiological Effluent Controls Program.

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 -----NOTE----- The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition. ----- Place CREATS in Mode F.	1 hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1 Initiate action to restore channel(s) to OPERABLE status.	Immediately
	<u>AND</u>	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.3 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform COT.	92 days
SR 3.3.6.2 -----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	24 months
SR 3.3.6.3 Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.4 Perform ACTUATION LOGIC TEST.	24 months

CREATS Actuation Instrumentation
3.3.6

Table 3.3.6-1 (page 1 of 1)
CREATS Actuation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1 train	SR 3.3.6.2	NA
2. Automatic Actuation Logic and Actuation Relays	1 train	SR 3.3.6.4	NA
3. Control Room Radiation Intake Monitor			
a. Iodine	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 9 \times 10^9$ $\mu\text{Ci/cc}$
b. Noble Gas	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 1 \times 10^6$ $\mu\text{Ci/cc}$
c. Particulate	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 1 \times 10^8$ $\mu\text{Ci/cc}$

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

Atomic Industry Forum (AIF) GDC 14 (Ref. 1) requires that the core protection systems, together with associated engineered safety features equipment, be designed to prevent or suppress conditions that could result in exceeding acceptable fuel design limits. The RTS initiates a plant shutdown, based on the values of selected plant parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The installed protection and monitoring systems have been designed to assure safe operation of the reactor at all times. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs with respect to these parameters and other reactor system parameters and equipment.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the associated LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs). These acceptable limits are:

- a. The Safety Limit (SL) values shall be maintained to prevent departure from nucleate boiling (DNB);
- b. Fuel centerline melt shall not occur; and
- c. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," maintains the above values and assures that offsite dose will be within 10 CFR 100 limits (Ref. 2) during AOOs.

(continued)

BASES

BACKGROUND
(continued)

DBAs are events that are analyzed even though they are not expected to occur during the plant life. The DBA acceptance limit is that offsite doses shall be maintained within an acceptable fraction of 10 CFR 100 limits (Ref. 2). There are five different accident categories which are organized based on the probability of occurrence (Ref. 3). Each accident category is allowed a different fraction of the 10 CFR 100 limits, inversely proportioned to the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered as having acceptable consequences for that event.

The RTS instrumentation is segmented into three distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 4):

- a. Field transmitters or process sensors;
- b. Signal process control and protection equipment; and
- c. Reactor trip switchgear.

These modules are shown in Figure B 3.3.1-1 and discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. To account for the calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor as provided in established plant procedures.

(continued)

BASES

BACKGROUND
(continued)

Signal Process Control and Protection Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 7 (Ref. 4), Chapter 6 (Ref. 5), and Chapter 15 (Ref. 6). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter has no measurable setpoint and is only used as an input to the protection circuits (e.g., manual trip functions) two channels with a one-out-of-two logic are sufficient. A third channel is not required since no surveillance testing is required during the time period in which the parameter is required.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 7).

(continued)

BASES

BACKGROUND

Signal Process Control and Protection Equipment (continued)

The two, three, and four process control channels discussed above all feed two logic trains. Figure B 3.3.1-1 shows a two-out-of-four logic function which provides input into two logic trains (Train A and B). Two logic trains are required to ensure that no single failure of one logic train will disable the RTS. Provisions to allow removing logic trains from service during maintenance are unnecessary because of the logic system's designed reliability. During normal operation, the two logic trains remain energized.

Reactor Trip Switchgear

The reactor trip switchgear includes the reactor trip breakers (RTBs) and bypass breakers as shown on Figure B 3.3.1-1. The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the control rod drive mechanisms (CRDMs). Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity and shutdown the reactor. Each RTB may be bypassed with a bypass breaker to allow testing of the RTB while the plant is at power. During normal operation, the output from the protection system is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the protection system output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open allowing the shutdown rods and control rods to fall into the core. Therefore, a loss of power to the protection system or RTBs will cause a reactor trip. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the protection system (except for the zirconium guide tube trip which only utilizes the undervoltage coils). Either the undervoltage coil or the shunt trip mechanism is sufficient by itself to open the RTBs, thus providing diverse trip mechanisms.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The RTS functions to maintain the SLs during all AOs and mitigates the consequences of DBAs which initiate in any MODE in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 6 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as anticipatory trips to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for a RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO and
APPLICABILITY
(continued)

The LCO and Applicability of each RTS Function are provided in Table 3.3.1-1. Included on Table 3.3.1-1 are Trip Setpoints for all applicable RTS Functions. Trip Setpoints for RTS Functions not specifically modeled in the safety analysis are based on established limits provided in the UFSAR (Reference 4). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS. The Trip Setpoints are the limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy as specified within plant procedures. The channel containing the bistable is considered inoperable when the "as found" value exceeds the Trip Setpoint specified in Table 3.3.1-1.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 4, 5, and 6. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.1-1 are therefore conservatively adjusted with respect to the analytical limits used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 8.

The RTS utilizes various permissive signals to ensure reactor trip Functions are in the correct configuration for the current plant status. These permissives back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Function is available.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY
(continued)

The safety analyses and OPERABILITY requirements applicable to each RTS Function and permissive provided in Table 3.3.1-1 are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip Function ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip pushbuttons on the main control board. A Manual Reactor Trip energizes the shunt trip device and de-energizes the undervoltage coils for the RTBs and bypass breakers. It is used at the discretion of the control room operators to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint or during other degrading plant conditions.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip pushbutton which actuates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single failure will disable the Manual Reactor Trip Function. This function has no adjustable trip setpoint with which to associate an LSSS, therefore no setpoints are provided.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY

1. Manual Reactor Trip (continued)

In MODE 1 or 2, manual initiation capability of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the RTBs are closed and the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip is not required to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods, or if one or more RTBs are open. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

2. Power Range Neutron Flux

The Power Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident. The Nuclear Instrumentation System (NIS) power range detectors (N-41, N-42, N-43, and N-44) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the CRD System for determination of automatic rod speed and direction. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires all four of the Power Range Neutron Flux-High trip Function channels to be OPERABLE.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

a. Power Range Neutron Flux-High (continued)

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux-High trip Function is not required to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low trip Function channels (N-41, N-42, N-43, and N-44) to be OPERABLE.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

b. Power Range Neutron Flux - Low (continued)

In MODE 1, below 6% RTP, and in MODE 2, the Power Range Neutron Flux - Low trip must be OPERABLE. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than approximately 8% RTP (P-10 setpoint). This Function is automatically unblocked when three-out-of-four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux - High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function is not required to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux - Low trip Function and is not specifically modeled in the accident analysis. The NIS intermediate range detectors (N-35 and N-36) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

(continued)

BASES

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3. Intermediate Range Neutron Flux (continued)

The LCO requires two channels of the Intermediate Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. Because this trip Function is important only during low power conditions, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below 6% RTP, and in MODE 2, the Intermediate Range Neutron Flux trip Function must be OPERABLE since there is a potential for an uncontrolled RCCA bank rod withdrawal accident. Above 8% RTP (P-10 setpoint), the Power Range Neutron Flux-High trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip Function is not required to be OPERABLE because the NIS intermediate range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against reactivity additions or power excursions in MODE 3, 4, 5, or 6.

(continued)

BASES

APPLICABLE
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(continued)

4. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition and provides protection against boron dilution and rod ejection events. This trip Function provides redundant protection to the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip Functions in MODE 2 and is not specifically credited in the accident analysis at these conditions. The NIS source range detectors (N-31 and N-32) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux trip Function to be OPERABLE in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods fully inserted. In this case, the source range Function is to provide control room indication. The outputs of the Function to RTS logic are not required to be OPERABLE when the CRD system is not capable of rod withdrawal and all rods fully inserted.

The Source Range Neutron Flux Trip Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

(continued)

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4. Source Range Neutron Flux (continued)

In MODE 2 when both intermediate range channels are $< 5E-11$ amps (below the P-6 setpoint), the Source Range Neutron Flux trip Function must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are manually de-energized by the operator and are inoperable.

In MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods are not fully inserted, the Source Range Neutron Flux trip Function must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal and all rods are fully inserted, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

(continued)

BASES

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(continued)

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit departure from nucleate boiling ratio (DNBR) is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, T_{avg} , axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow.

Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses the ΔT of each loop as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution $f(\Delta I)$ — the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

(continued)

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5. Overtemperature ΔT (continued)

The Overtemperature ΔT trip Function is calculated in two channels for each loop as described in Note 1 of Table 3.3.1-1. A reactor trip occurs if the Overtemperature ΔT Trip Setpoint is reached in two-out-of-four channels. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent an unnecessary reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

(continued)

BASES

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(continued)

6. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system; and
- axial power distribution $f(\Delta I)$ — the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

(continued)

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6. Overpower ΔT (continued)

The Overpower ΔT trip Function is calculated in two channels for each loop as described in Note 2 of Table 3.3.1-1. A reactor trip occurs if the Overpower ΔT trip setpoint is reached in two-out-of-four channels. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent an unnecessary reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only MODES where enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

(continued)

BASES

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(continued)

7. Pressurizer Pressure

The same sensors (PT-429, PT-430, and PT-431) provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip with the exception that the Pressurizer Pressure-Low and Overtemperature ΔT trips also receive input from PT-449. Since the Pressurizer Pressure channels are also used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function.

a. Pressurizer Pressure - Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The LCO requires four channels of the Pressurizer Pressure-Low trip Function to be OPERABLE. Included within the four channels are lead time and lead/lag constraints.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip function must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (8.5% RTP). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, the Pressurizer Pressure-Low trip Function is not required to be OPERABLE because no conceivable power distributions can occur that would cause DNB concerns.

(continued)

BASES

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(continued)

b. Pressurizer Pressure - High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions. The LCO requires three channels of the Pressurizer Pressure-High trip Function to be OPERABLE.

In MODE 1 or 2, the Pressurizer Pressure-High trip Function must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function is not required to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when in or below MODE 4.

8. Pressurizer Water Level - High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. This trip Function is not specifically modeled in the accident analysis.

(continued)

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8. Pressurizer Water Level-High (continued)

The LCO requires three channels of the Pressurizer Water Level-High trip Function to be OPERABLE. The pressurizer level channels (LT-426, LT-427, and LT-428) are also used for other control functions. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before the reactor high pressure trip.

In MODE 1 or 2, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip Function must be OPERABLE. In MODES 3, 4, 5, or 6, the Pressurizer Water Level-High trip Function is not required to be OPERABLE because transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

9. Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low (Single Loop) and (Two Loops) trip Functions utilize three common flow transmitters per RCS loop to generate a reactor trip above 8.5% RTP (P-7 setpoint). Flow transmitters FT-411, FT-412, and FT-413 are used for RCS Loop A and FT-414, FT-415, and FT-416 are used for RCS Loop B.

(continued)

BASES

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(continued)

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in the RCS loop, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, (50% RTP), a loss of flow in either RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low (Single Loop) trip Function channels per RCS loop to be OPERABLE in MODE 1 \geq 50% RTP (above P-8 setpoint). Each loop is considered a separate Function for the purpose of this LCO.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint the Reactor Coolant Flow-Low (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in both RCS loops while avoiding reactor trips due to normal variations in loop flow.

The LCO requires three Reactor Coolant Flow-Low (Two Loops) trip Function channels per loop to be OPERABLE in MODE 1 above 8.5% RTP (P-7 setpoint) and before the Reactor Coolant Flow-Low (Single Loop) trip Function is OPERABLE (below the P-8 setpoint). Each loop is considered a separate Function for the purpose of this LCO.

(continued)

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b. Reactor Coolant Flow-Low (Two Loops)
(continued)

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in both loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

Below the P-7 setpoint, this trip Function is not required to be OPERABLE because all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip Function is not required to be OPERABLE because loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

10. RCP Breaker Position

Both RCP Breaker Position trip Functions (Single Loop and Two Loops) utilize a common auxiliary contact located on each RCP. These Functions anticipate the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates but are not specifically credited in the accident analysis.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above 50% RTP, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Single Loop) Trip Setpoint is reached.

(continued)

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a. RCP Breaker Position (Single Loop) (continued)

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 \geq 50% RTP (above the P-8 setpoint). Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of plant SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip Function must be OPERABLE. In MODE 1 below the P-8 setpoint, the RCP Breaker Position (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

b. RCP Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops. The position of each RCP breaker is monitored. If both RCP breakers are open above 8.5% RTP (P-7 setpoint) and before the RCP Breaker Position (Single Loop) trip Function is OPERABLE (below the P-8 setpoint), a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached.

(continued)

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b. Reactor Coolant Pump Breaker Position (Two Loops)
(continued)

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 above the P-7 and below the P-8 setpoints. Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this Function because the RCS Flow-Low trip alone provides sufficient protection of plant SIs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow (including RCP breaker position) are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function is not required to be OPERABLE because a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

BASES

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(continued)

11. Undervoltage - Bus 11A and 11B

The Undervoltage - Bus 11A and 11B reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops from a major network voltage disturbance. The voltage to each RCP is monitored. Above 8.5% RTP (the P-7 setpoint), an undervoltage condition detected on both Buses 11A and 11B will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage Bus 11A and 11B channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Undervoltage - Bus 11A and 11B trip Function channels per bus to be OPERABLE in MODE 1 above the P-7 setpoint. Each bus is considered a separate Function for the purpose of this LCO.

Below the P-7 setpoint, the Undervoltage - Bus 11A and 11B trip Function is not required to be OPERABLE because all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Undervoltage - Bus 11A and 11B is automatically enabled.

(continued)

BASES

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APPLICABILITY
(continued)

12. Underfrequency - Bus 11A and 11B

The Underfrequency - Bus 11A and 11B reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCP loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above 8.5% RTP (the P-7 setpoint), a loss of frequency detected on both RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Underfrequency - Bus 11A and 11B channels per bus to be OPERABLE in Mode 1 above the P-7 setpoint. Each bus is considered a separate Function for the purpose of this LCO.

Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Underfrequency - Bus 11A and 11B is automatically enabled.

(continued)

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13. Steam Generator Water Level - Low Low

The Steam Generator (SG) Water Level - Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. Three level transmitters per SG (LT-461, LT-462, and LT-463 for SG A and, LT-471, LT-472, and LT-473 for SG B) provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the Engineered Safety Feature Actuation System (ESFAS) function of starting the AFW pumps on low low SG level. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor.

The LCO requires three trip Function channels of SG Water Level - Low Low per SG to be OPERABLE in MODES 1 and 2. Each SG is considered a separate Function for the purpose of this LCO.

In MODE 1 or 2, the SG Water Level - Low Low trip Function must be OPERABLE to ensure that a heat sink is available to the reactor. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low trip Function is not required to be OPERABLE because the reactor is not operating. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

(continued)

BASES

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(continued)

14. Turbine Trip

Credit for these trip Functions is not credited in the accident analysis.

a. Turbine Trip-Low Autostop Oil Pressure

The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint. Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. Three pressure switches monitor the control oil pressure in the Autostop Oil System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three trip Function channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, the Turbine Trip-Low Autostop Oil Pressure trip Function is not required to be OPERABLE because load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore, there is no potential for a turbine trip.

(continued)

BASES

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(continued)

b. Turbine Trip - Turbine Stop Valve Closure

The Turbine Trip - Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint. Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip - Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If both limit switches indicate that the stop valves are closed, a reactor trip is initiated.

This Function only measures the discrete position (open or closed) of the turbine stop valves. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

The LCO requires two Turbine Trip - Turbine Stop Valve Closure trip Function channels, one per valve, to be OPERABLE in MODE 1 above P-9. Both channels must trip to cause reactor trip.

(continued)

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APPLICABILITY

b. Turbine Trip - Turbine Stop Valve Closure
(continued)

Below the P-9 setpoint, the Turbine Trip - Turbine Stop Valve Closure trip Function is not required to be OPERABLE because a load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore there is no potential for a turbine trip.

15. Safety Injection Input from Engineered Safety Feature Actuation System

The Safety Injection (SI) Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This trip is assumed in the safety analyses for the loss of coolant accident (LOCA). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoints are not applicable to this Function. The SI Input is provided by relays in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trip Function channels of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

(continued)

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SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

16. Reactor Trip System Interlocks

Reactor protection interlocks (i.e., permissives) are provided to ensure reactor trips are in the correct configuration for the current plant status. They back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6 Permissive

The Intermediate Range Neutron Flux, P-6 permissive is actuated when any NIS intermediate range channel goes approximately one decade ($1 \text{ E-}10$ amps) above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 permissive ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip by use of two defeat push buttons. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the Source Range Neutron Flux reactor trip at $5\text{E-}11$ amps.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

a. Intermediate Range Neutron Flux, P-6 Permissive
(continued)

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 permissive to be OPERABLE in MODE 2 when below the P-6 permissive setpoint.

Above the P-6 permissive setpoint, the Source Range Neutron Flux reactor trip will be blocked, and this Function is no longer required.

In MODE 3, 4, 5, or 6 the P-6 permissive does not have to be OPERABLE because the Source Range is providing the required core protection.

b. Low Power Reactor Trips Block, P-7 Permissive

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or from first stage turbine pressure. The LCO requirement for the P-7 permissive allows the bypass of the following Functions:

- Pressurizer Pressure - Low;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage - Bus 11A and 11B; and
- Underfrequency - Bus 11A and 11B.

These reactor trip functions are not required below the P-7 setpoint since the RCS is capable of providing sufficient natural circulation without any RCP running.

The LCO requires four channels of Low Power Reactor Trips Block, P-7 permissive to be OPERABLE in MODE 1 \geq 8.5% RTP.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

b. Low Power Reactor Trips Block, P-7 Permissive
(continued)

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the permissive performs its Function when power level drops below 8.5% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8 Permissive

The Power Range Neutron Flux, P-8, permissive is actuated at approximately 49% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock allows the Reactor Coolant Flow-Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on low flow in one or more RCS loops to be blocked so that a loss of a single loop will not cause a reactor trip. The LCO requirement for this trip Functions ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when $\geq 50\%$ power.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1 $\geq 50\%$ RTP.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 permissive must be OPERABLE. In MODE 1 $< 50\%$ RTP, this function is not required to be OPERABLE because a loss of flow in one loop will not result in DNB. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

d. Power Range Neutron Flux, P-9 Permissive

The Power Range Neutron Flux, P-9 permissive is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors if the Steam Dump System is available and at 8% if the Steam Dump System is unavailable. The LCO requirement for this Function ensures that the Turbine Trip-Low Autostop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System and RCS. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO require four channels of Power Range Neutron Flux, P-9 permissive to be OPERABLE in MODE 1 above the permissive setpoint.

In MODE 1 above the permissive setpoint, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System and RCS, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 1 below the permissive setpoint and MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10 Permissive

The Power Range Neutron Flux, P-10 permissive is actuated at approximately 8% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 8% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 permissive ensures that the following Functions are performed:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

- e. Power Range Neutron Flux, P-10 Permissive
(continued)
- on increasing power, the P-10 permissive allows the operator to manually block the Intermediate Range Neutron Flux and Power Range Neutron Flux-low reactor trips;
 - on increasing power, the P-10 permissive automatically provides a backup signal to the P-6 permissive to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detector;
 - the P-10 interlock provides one of the two inputs to the P-7 interlock; and
 - on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 < 6% RTP and MODE 2.

OPERABILITY in MODE 1 < 6% RTP ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must also be OPERABLE in MODE 2 to ensure that core protection is providing during a startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

17. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The OPERABILITY requirement for the individual trip mechanisms is provided in Function 18 below. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the CRD System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the CRD System, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

19. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is also equipped with a redundant bypass breaker to allow testing of the trip breaker while the plant is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE trains ensures that failure of a single logic train will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

The RTS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to analytical values specified in plant procedures, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

(continued)

BASES

ACTIONS
(continued)

As shown on Figure B 3.3.1-1, the RTS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Trip Logic (Function 19 in Table 3.3.1-1). Therefore, a channel may be inoperable due to the failure of a field instrument or a bistable failure which affects one or both RTS trains that is comprised of the RTBs and Automatic Trip Logic Function. The only exception to this are the Manual Reactor Trip and SI Input from ESFAS trip Functions which are defined strictly on a train basis (i.e., failure of these Functions may only affect one RTS train).

A.1

Condition A applies to all RTS protection functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable or if both source range channels are inoperable. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in a trip Function exceed those specified in all related Conditions associated with a trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if the trip Function is applicable in the current MODE of operation. This essentially applies to the loss of more than one channel of any RTS Function except with respect to Conditions G and H.

B.1.

Condition B applies to the Manual Reactor Trip Function in MODE 1 or 2 and in MODES 3, 4, and 5 with the CRD system capable of rod withdrawal or all rods not fully inserted. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the required safety function.

(continued)

BASES

ACTIONS

B.1 (continued)

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

C.1, C.2, and C.3

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time of Condition B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours, action must be initiated within 6 hours to ensure that all rods are fully inserted, and the Control Rod Drive System must be placed in a condition incapable of rod withdrawal within 7 hours. The Completion Times provide adequate time to exit the MODE of Applicability from full power operation in an orderly manner without challenging plant systems based on operating experience.

D.1

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High; and
- SG Water Level-Low Low.

(continued)

BASES

ACTIONS

D.1 (continued)

With one channel inoperable, the channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition. For the Power Range Neutron Flux-High, Power Range Neutron Flux-Low, Overtemperature ΔT , and Overpower ΔT functions, this results in a one-out-of-three logic for actuation. For the Pressurizer Pressure-High and Pressurizer Water Level-High Functions, this results in a one-out-of-two logic for actuation. For the SG Water Level-Low Low Function, this results in a one-out-of-two logic per each affected SG for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing surveillance testing of other channels. This includes placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. This 4 hours is applied to each of the remaining OPERABLE channels. The 4 hour time limit is consistent with Reference 9.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

Condition E applies to the Intermediate Range Neutron Flux trip Function when THERMAL POWER is above the P-6 setpoint (5E-11 amp as derived from a bistable circuit of the intermediate range channels) and below the P-10 setpoint (6% RTP as derived from a bistable circuit of the Power Range channels) and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs a monitoring and protection function. With one NIS intermediate range channel inoperable, 2 hours is allowed to either reduce THERMAL POWER below the P-6 setpoint or increase THERMAL POWER above the P-10 setpoint. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel inoperability does not result in reactor trip.

Required Action E.2 is modified by a Note which states that the option to increase THERMAL POWER is not allowed if both intermediate range channels are inoperable or if THERMAL POWER is < 5E-11 amps. This prevents the plant from increasing THERMAL POWER when the trip capability of the Intermediate Range Neutron Flux trip Function is not available or if the plant has not yet entered this trip Function's MODE of Applicability.

(continued)

BASES

ACTIONS
(continued)

F.1, F.2, and F.3

Condition F applies to the Source Range Neutron Flux trip Function when in MODE 2, below the P-6 setpoint. In this Condition, the NIS source range performs the monitoring and protection functions. With two channels inoperable, the RTBs and RTBBs must be opened immediately. With the RTBs and RTBBs opened, the core is in a more stable condition.

With one channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation since with only one source range channel OPERABLE, core protection is severely reduced. The inoperable channel must also be restored within 48 hours.

G.1

If the Required Actions of Condition D, E, or F cannot be met within the specified Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

H.1, H.2, and H.3

Condition H applies to an inoperable source range channel in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted. In this Condition, the NIS source range performs the monitoring and protection functions. With two channels inoperable, at least one channel must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this interval.

(continued)

BASES

ACTIONS
(continued)

H.1, H.2, and H.3

With one of the source range channels inoperable, operations involving positive reactivity additions must be suspended immediately and 48 hours is allowed to restore it to OPERABLE status. The suspension of positive reactivity additions will preclude any power escalation.

I.1 and I.2

If the Source Range trip Function cannot be restored to OPERABLE status within the required Completion Time of Condition H, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, action must be immediately initiated to fully insert all rods. Additionally, the CRD System must be placed in a condition incapable of rod withdrawal within 1 hour. The Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event occurring during this interval.

J.1

Condition J applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods are fully inserted. In this Condition, the NIS source range performs the monitoring function. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

(continued)

BASES

ACTIONS

J.1 (continued)

Also, the SDM must be verified once within 12 hours and every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM once per 12 hours allows sufficient time to perform the calculations and determine that the SDM requirements are met and to ensure that the core reactivity has not changed. Required Action J.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Time of once per 12 hours is based on operating experience in performing the Required Actions and the knowledge that plant conditions will change slowly.

K.1

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage - Bus 11A and 11B; and
- Underfrequency - Bus 11A and 11B.

With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 6 hours allowed to place the channel in the tripped condition is consistent with Reference 9 if the inoperable channel cannot be restored to OPERABLE status.

(continued)

BASES

ACTIONS

K.1 (continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel(s), and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

For the Reactor Coolant Flow-Low (Two Loops) Function, Condition K applies on a per loop basis. For the RCP Breaker Position (Two Loops) Function, Condition K applies on a per RCP basis. For Undervoltage-Bus 11A and 11B and underfrequency-Bus 11A and 11B, Condition K applies on a per bus basis. This allows one inoperable channel from each loop, RCP, or bus to be considered on a separate condition entry basis.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hour time limit is consistent with Reference 9. The 4 hours is applied to each of the remaining OPERABLE channels.

L.1

If the Required Action and Completion Time of Condition K is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 1 < 8.5% RTP (P-7 setpoint) at which point the Function is no longer required. An alternative is not provided for increasing THERMAL POWER above the P-8 setpoint for the Reactor Coolant Flow-Low (Two Loops) and RCP Breaker Position (Two Loops) trip Functions since this places the plant in Condition M. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

M.1

Condition M applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel per loop inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status or place in trip is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hours is applied to each of the two OPERABLE channels. The 4 hour time limit is consistent with Reference 9.

N.1

Condition N applies to the RCP Breaker Position (Single Loop) trip Function. There is one breaker position device per RCP breaker. With one channel per RCP inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status is consistent with Reference 9.

O.1

If the Required Action and associated Completion Time of Condition M or N is not met, the plant must be placed in a MODE where the Functions are not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-8 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

(continued)

BASES

ACTIONS
(continued)

P.1

Condition P applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure in MODE 1 above the P-9 setpoint. With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. If placed in the tripped Condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 6 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hours is applied to each remaining OPERABLE channel. The 4 hour time limit is consistent with Reference 9.

Q.1, Q.2.1, and Q.2.2

If the Required Action and Associated Completion Time of Condition P are not met, the plant must be placed in a MODE where the Turbine Trip Functions are no longer required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-9 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

The Steam Dump system must also be verified OPERABLE within 7 hours or THERMAL POWER must be reduced to < 8% RTP. This ensures that either the secondary system or RCS is capable of handling the heat rejection following a reactor trip. The Completion Times are reasonable considering the need to perform the actions in an orderly manner and the low probability of an event occurring in this time.

(continued)

BASES

ACTIONS
(continued)

R.1

Condition R applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. With one train inoperable, 6 hours is allowed to restore the train to OPERABLE status. The Completion Time of 6 hours to restore the train to OPERABLE status is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval.

The Required Action has been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

S.1 and S.2

Condition S applies to the P-6, P-7, P-8, P-9, and P-10 permissives. With one channel inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour or the associated RTS channel(s) must be declared inoperable. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions.

T.1

Condition T applies to the RTBs in MODES 1 and 2. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status. The 1 hour Completion Time is based on operating experience and the minimum amount of time allowed for manual operator actions.

The Required Action has been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE.

(continued)

BASES

ACTIONS
(continued)

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms (i.e., diverse trip features) in MODES 1 and 2. With two diverse trip features inoperable, at least one diverse trip feature must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism for one RTB inoperable, it must be restored to an OPERABLE status within 48 hours. The affected RTB shall not be bypassed while one of the diverse trip features is inoperable except for the time required to perform maintenance to one of the diverse trip features. The allowable time for performing maintenance of the diverse trip features is 6 hours for the reasons stated under Condition T. The Completion Time of 48 hours for Required Action U.2 is reasonable considering that in this Condition there is one remaining diverse trip feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1

If the Required Action and Associated Completion Time of Condition R, S, T, or U is not met, the plant must be placed in a MODE where the Functions are no longer required to be OPERABLE. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

W.1 and W.2

Condition W applies to the following reactor trip Functions in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted:

- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

With two trip mechanisms inoperable, at least one trip mechanism must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism or train inoperable, the inoperable trip mechanism or train must be restored to OPERABLE status within 48 hours.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

X.1 and X.2

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action must be initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies (Ref. 8).

SR 3.3.1.1

A CHANNEL CHECK is required for the following RTS trip functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure-Low;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);
- Reactor Coolant Flow-Low (Two Loops); and

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1 (continued)

- SG Water Level - Low Low

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel check acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

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REQUIREMENTS

SR 3.3.1.2

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux-High channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2 (continued)

This SR is modified by a Note which states that this Surveillance is required only if reactor power is $\geq 50\%$ RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

This SR compares the incore system to the NIS channel output every 31 effective full power days (EFPD). If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the Overtemperature ΔT Function.

This SR is modified by two Notes. Note 1 clarifies that the Surveillance is required to be performed within 7 days after THERMAL POWER is $\geq 50\%$ RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 31 EFPD. Note 2 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.3 (continued)

The Frequency of every 31 EFPD is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

This SR is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS of the RTB, and the RTB Undervoltage and Shunt Trip Mechanisms. This test shall verify OPERABILITY by actuation of the end devices.

The test shall include separate verification of the undervoltage and shunt trip mechanisms except for the bypass breakers which do not require separate verification since no capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.11. However, the bypass breaker test shall include a local shunt trip. This test must be performed on the bypass breaker prior to placing it in service to take the place of a RTP.

The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

This SR is the performance of an ACTUATION LOGIC TEST on the RTS Automatic Trip Logic every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.6

This SR is a calibration of the excore channels to the incore channels every 92 EFPD. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the Overtemperature ΔT Function.

This SR has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is $\geq 50\%$ RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 92 EFPD.

The Frequency of 92 EFPD is adequate based on industry operating experience; considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

This SR is the performance of a COT every 92 days for the following RTS functions:

- Power Range Neutron Flux - High;
- Source Range Neutron Flux (in MODE 3, 4, or 5 with CRD System capable of rod withdrawal or all rods not fully inserted);
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure - Low;
- Pressurizer Pressurizer - High;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Single Loop);

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.7 (continued)

- Reactor Coolant Flow-Low (Two Loops); and
- SG Water Level-Low Low

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Trip Setpoint of Table 3.3.1-1. The "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8).

This SR is modified by a Note that provides a 4 hour delay in the requirement to perform this surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the plant is in MODE 3 with the RTBs closed for greater than 4 hours, this SR must be performed within 4 hours after entry into MODE 3.

The Frequency of 92 days is consistent with Reference 9.

SR 3.3.1.8

This SR is the performance of a COT as described in SR 3.3.1.7 for the Power Range Neutron Flux-Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux (MODE 2), except that this test also includes verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition. This SR is modified by two Notes that provide a 4 hour delay in the requirement to perform this surveillance. These Notes allow a normal shutdown to be completed and the plant removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 4 hours after reducing power below P-10 or P-6.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.8 (continued)

The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the Source range channels. Once the plant is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the plant in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

This SR is the performance of a TADOT for the Undervoltage-Bus 11A and 11B and Underfrequency-Bus 11A and 11B trip Functions. The Frequency of every 92 days is consistent with Reference 9.

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to Bus 11A and 11B undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION required by SR 3.3.1.10.

SR 3.3.1.10

This SR is the performance of a CHANNEL CALIBRATION for the following RTS Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.10 (continued)

- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure - Low;
- Pressurizer Pressure - High;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Single Loop);
- Reactor Coolant Flow - Low (Two Loops);
- Undervoltage - Bus 11A and 11B;
- Underfrequency - Bus 11A and 11B;
- SG Water Level - Low Low;
- Turbine Trip - Low Autostop Oil Pressure; and
- Reactor Trip System Interlocks.

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology (Ref. 8). The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of 24 month calibration intervals in the determination of the magnitude of equipment drift in the setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.10 (continued)

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors shall include an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 50% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.1.11

This SR is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS trip Functions. This TADOT is performed every 24 months. This test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.11 (continued)

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT because the Functions affected have no setpoints associated with them.

SR 3.3.1.12

This SR is the performance of a TADOT for Turbine Trip Functions which is performed prior to reactor startup if it has not been performed within the last 31 days. This test shall verify OPERABILITY by actuation of the end devices.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

This SR is modified by a Note stating that verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical because this test cannot be performed with the reactor at power.

SR 3.3.1.13

This SR is the performance of a COT of the RTS interlocks every 24 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)



BASES (continued)

- REFERENCES
1. Atomic Industry Forum (AIF) GDC 14, Issued for comment July 10, 1967.
 2. 10 CFR 100.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. UFSAR, Chapter 7.
 5. UFSAR, Chapter 6.
 6. UFSAR, Chapter 15.
 7. IEEE-279-1971.
 8. RG&E Engineering Work Request (EWR) 5126, "Guidelines for Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
 9. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
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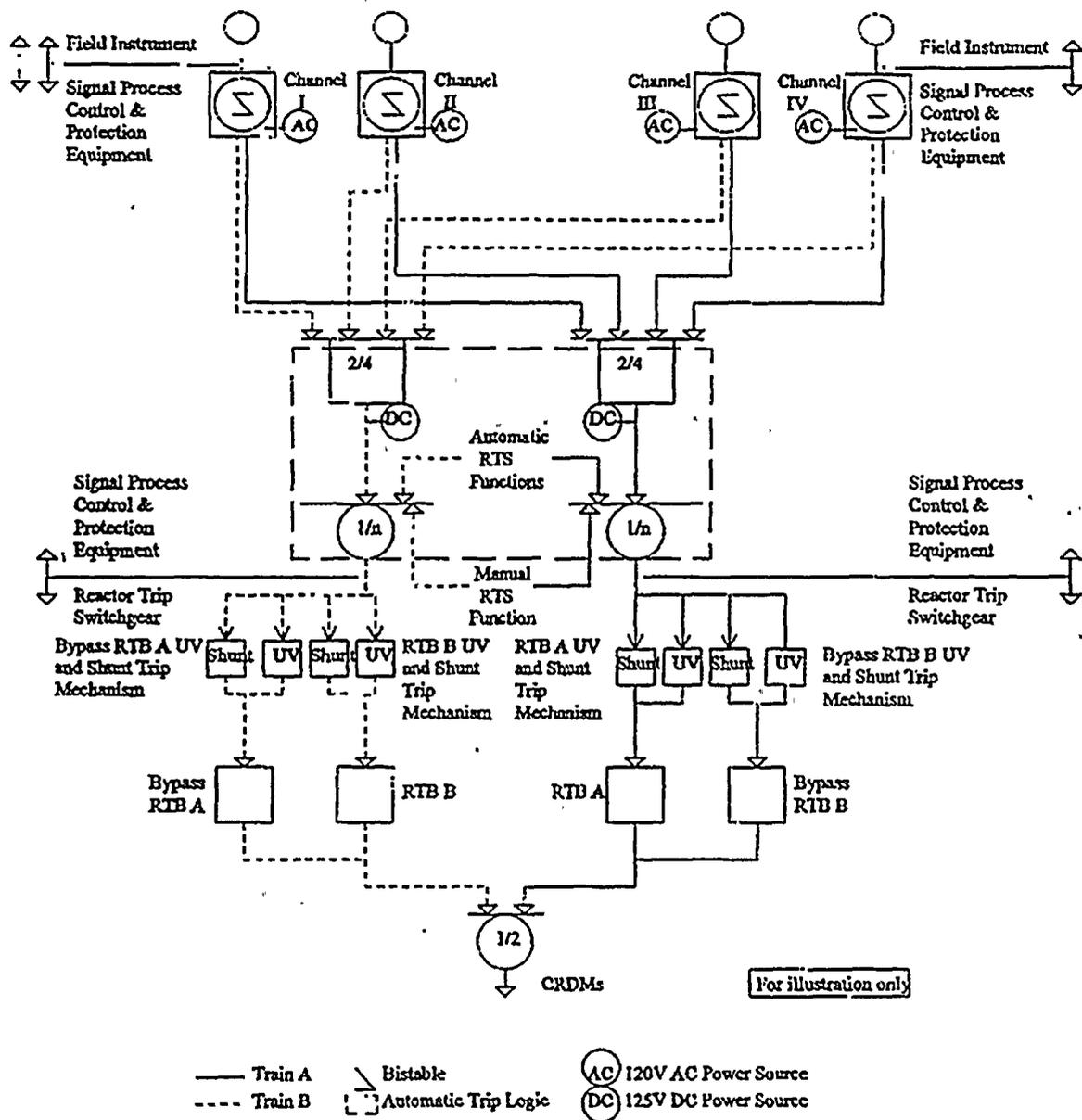


Figure B 3.3.1-1

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

Atomic Industrial Forum (AIF) GDC 15 (Ref. 1) requires that protection systems be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

The ESFAS initiates necessary safety systems, based on the values of selected plant parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into two distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 2):

- Field transmitters or process sensors; and
- Signal processing equipment.

These modules are discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). To account for calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor.

(continued)

BASES

BACKGROUND
(continued)

Signal Processing Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 6 (Ref. 3), Chapter 7 (Ref. 2), and Chapter 15 (Ref. 4). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 5).

The actuation of ESF components is accomplished through master and slave relays. The protection system energizes the master relays appropriate for the condition of the plant. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices.

(continued)

BASES (continued)

APPLICABLE
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LCO, AND
APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, SI-Pressurizer Pressure-Low is a primary actuation signal for small break loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as anticipatory actions to Functions that were credited in the accident analysis (Ref. 4).

This LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single failure disables the ESFAS.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The LCO and Applicability of each ESFAS Function are provided in Table 3.3.2-1. Included on Table 3.3.2-1 are Allowable Values and Trip Setpoints for all applicable ESFAS Functions. Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated within the LCOs, including any Required Actions that are in effect at the onset of the DBA and the equipment functions as designed.

The Trip Setpoints are the limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 2, 3, and 4. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.2-1 are therefore conservatively adjusted with respect to the analytical limits (i.e., Allowable Values) used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 6. If the measured setpoint exceeds the Trip Setpoint Value, the bistable is considered OPERABLE unless the Allowable Value as specified in plant procedures is exceeded. The Allowable Value specified in the plant procedures bounds that provided in Table 3.3.2-1 since the values in the table are typically those used in the accident analysis.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 have been confirmed based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protection functions provided in Table 3.3.2-1 are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to $< 2200^{\circ}\text{F}$); and
2. Boration to ensure recovery and maintenance of SDM ($k_{\text{eff}} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Containment Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;
- Feedwater Isolation; and
- Start of motor driven auxiliary feedwater (AFW) pumps.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Safety Injection (continued)

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses; and
- Start of AFW to ensure secondary side cooling capability.

a. Safety Injection - Manual Initiation

This LCO requires one channel per train to be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. The operator can initiate SI at any time by using either of two pushbuttons on the main control board. This action will cause actuation of all components with the exception of Containment Isolation and Containment Ventilation Isolation.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one pushbutton and the interconnecting wiring to the actuation logic cabinet. Each pushbutton actuates both trains. This configuration does not allow testing at power.

(continued)

BASES

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a. Safety Injection - Manual Initiation (continued)

This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems. Also, this Function is not required in MODE 4 since it does not actuate Containment Isolation or Containment Ventilation Isolation.

b. Safety Injection - Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE in MODES 1, 2, 3, and 4. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

This Function is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters and electronics are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure-High must be OPERABLE in MODES 1, 2, 3, and 4 because there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 5 and 6, Containment Pressure-High is not required to be OPERABLE because there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) atmospheric relief or safety valve;
- SLB;

(continued)

BASES

APPLICABLE
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d. Safety Injection - Pressurizer Pressure - Low
(continued)

- Rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Since there are dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the interlock setpoint. Automatic SI actuation below this interlock setpoint is performed by the Containment Pressure-High signal.

This Function is not required to be OPERABLE in MODE 3 below the Pressurizer Pressure interlock setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO; and
APPLICABILITY
(continued)

e. Safety Injection - Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG atmospheric relief or an SG safety valve.

Steam line pressure transmitters provide control input, but the control function cannot initiate events that the Function acts to mitigate. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line. Each steam line is considered a separate function for the purpose of this LCO.

With the transmitters located in the Intermediate Building, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

e. Safety Injection - Steam Line Pressure - Low
(continued)

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) when a secondary side break or stuck open SG atmospheric relief or safety valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the interlock setpoint. Below the interlock setpoint, a feed line break is not a concern. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the plant to cause an accident.

2. Containment Spray (CS)

CS provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and
3. Adjusts the pH of the water in containment sump B after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

(continued)

BASES

APPLICABLE
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LCO, and
APPLICABILITY

2. CS (continued)

CS is actuated manually or by Containment Pressure-High High. The CS actuation signal starts the CS pumps and aligns the discharge of the pumps to the CS nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the CS pumps and mixed with a sodium hydroxide solution from the spray additive tank. During the recirculation phase of accident recovery, the spray pump suctions are manually shifted to containment sump B if continued CS is required.

a. CS-Manual Initiation

The operator can initiate CS at any time from the control room by simultaneously depressing two CS actuation pushbuttons. Because an inadvertent actuation of CS could have serious consequences, two pushbuttons must be simultaneously depressed to initiate both trains of CS. Therefore, the inoperability of either pushbutton fails both trains of manual initiation.

Manual initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

b. CS - Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

c. CS - Containment Pressure - High High

This signal provides protection against a LOCA or an SLB inside containment. The transmitters are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is the only ESFAS Function that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate CS, since the consequences of an inadvertent actuation of CS could be serious.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

c. CS - Containment Pressure - High High
(continued)

The Containment Pressure - High High instrument function consists of two sets with three channels in each set. Each set is a two-out-of-three logic where the outputs are combined so that both sets tripped initiates CS. Each set is considered a separate function for the purposes of this LCO. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure - High High must be OPERABLE in MODES 1, 2, 3 and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and selected process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a LOCA.

(continued)

BASES

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3. Containment Isolation (continued)

Containment Isolation signals isolate all automatically isolatable process lines, except feedwater lines, main steam lines, and component cooling water (CCW). The main feedwater and steam lines are isolated by other functions since forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW enhances plant safety by allowing operators to use forced RCS circulation to cool the plant. Isolating CCW may require the use of feed and bleed cooling, which could prove more difficult to control.

a. Containment Isolation - Manual Initiation

Manual Containment Isolation is actuated by either of two pushbuttons on the main control board. Either pushbutton actuates both trains. Manual initiation of Containment Isolation also actuates Containment Ventilation Isolation.

Manual initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate plant conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

b. Containment Isolation - Automatic Actuation
Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate plant conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

c. Containment Isolation - Safety Injection

Containment Isolation is also initiated by all Functions that automatically initiate SI. The Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating Functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Closure of the main steam isolation valves (MSIVs) and their associated non-return check valves limits the accident to the blowdown from only the affected SG. For a SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

(continued)

BASES

APPLICABLE
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(continued)

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two actuation devices (one pushbutton and one switch) on the main control board for each MSIV. Each device can initiate action to immediately close its respective MSIV. The LCO requires one channel (device) per loop to be OPERABLE. Each loop is not considered a separate function since there is only one required per loop.

Manual initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

(continued)

BASES

APPLICABLE
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b. Steam Line Isolation - Automatic Actuation Logic
and Actuation Relays (continued)

Automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High High

This Function actuates closure of both MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters are located outside containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties. Containment Pressure-High High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic.

(continued)

BASES

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c. Steam Line Isolation - Containment
Pressure - High High (continued)

Containment Pressure - High High must be OPERABLE in MODES 1, 2, and 3, because there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The steam line isolation Function must be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6 the steam line isolation Function is not required to be OPERABLE because there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure - High High setpoint.

d. Steam Line Isolation - High Steam Flow Coincident
With Safety Injection and Coincident With
T_{avg} - Low

This Function provides closure of the MSIVs during an SLB or inadvertent opening of an SG atmospheric relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues. The one-out-of-two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.

(continued)

BASES

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- d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low (continued)

With the transmitters (d/p cells) located inside containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

Two channels of T_{avg} per loop are required to be OPERABLE for this Function. Each loop is considered a separate Function for the purpose of this LCO. The T_{avg} channels are combined in a logic such that any two of the four T_{avg} channels tripped in conjunction with SI and one of the two high steam line flow channels tripped causes isolation of the steam line associated with the tripped steam line flow channels. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels per loop in a two-out-of-four configuration ensures no single failure disables the T_{avg} - Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

BASES

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APPLICABILITY

- d. Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

- e. Steam Line Isolation - High High Steam Flow Coincident With Safety Injection

This Function provides closure of the MSIVs during a steam line break (or inadvertent opening of an SG atmospheric relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high-high steam flow in one steam line. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

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e. Steam Line Isolation - High High Steam Flow
Coincident With Safety Injection (continued)

The main steam lines isolate only if the high-high steam flow signal occurs coincident with an SI signal. Steamline isolation occurs only for the steam line associated with the tripped steam flow channels. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIV's are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

(continued)

BASES

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(continued)

5. Feedwater Isolation

The primary function of the Feedwater Isolation signals is to prevent and mitigate the effects of high water level in the SGs which could cause carryover of water into the steam lines and result in excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

This Function is actuated by either a SG Water Level-High or an SI signal. The Function provides feedwater isolation by closing the Main Feedwater Regulating Valves (MFRVs) and the associated bypass valves. In addition, on an SI signal, the AFW System is automatically started, and the MFW pump breakers are opened which closes the MFW pump discharge valves. The SI signal was discussed previously.

a. Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)



BASES

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(continued)

b. Feedwater Isolation - Steam Generator Water Level - High

The Steam Generator Water Level - High Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - High is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

c. Feedwater Isolation - Safety Injection

The Safety Injection Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)



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c. Feedwater Isolation - Safety Injection
(continued)

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The preferred system has two motor driven pumps and a turbine driven pump, making it available during normal plant operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break (depending on break location). A Standby AFW (SAFW) is also available in the event the preferred system is unavailable. The normal source of water for the AFW System is the condensate storage tank (CST) which is not safety related. Upon a low level in the CST the operators can manually realign the pump suctions to the Service Water (SW) System which is the safety related water source. The SW System also is the safety related water source for the SAFW System. The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately while the SAFW System is only manually initiated and aligned.

a. Auxiliary Feedwater - Manual Initiation

The operator can initiate AFW or SAFW at any time by using control switches on the Main Control board (one switch for each pump in each system). This action will cause actuation of their respective pump.

(continued)



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a. Auxiliary Feedwater - Manual Initiation
(continued)

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained to ensure the operator has manual AFW and SAFW initiation capability.

The LCO requires one channel per pump in each system to be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

b. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

(continued)



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b. Auxiliary Feedwater - Automatic Actuation Logic
and Actuation Relays (continued)

Automatic initiation of Auxiliary Feedwater must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

c. Auxiliary Feedwater - Steam Generator Water
Level - Low Low

SG Water Level - Low Low must be OPERABLE in MODES 1, 2, and 3 to provide protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level - Low Low in either SG will cause both motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level - Low Low in both SGs will cause the turbine driven pump to start. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

(continued)



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c. Auxiliary Feedwater - Steam Generator Water
Level - Low Low (continued)

Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - Low Low is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

One train of SG Water Level - Low Low channels are powered from Instrument Bus D. Therefore, if Instrument Bus D is inoperable, one train of Automatic Actuation Logic and Relays should be declared inoperable.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

d. Auxiliary Feedwater - Safety Injection

The SI function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

(continued)



BASES

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(continued)

e. Auxiliary Feedwater - Undervoltage - Bus 11A and 11B

The Undervoltage - Bus 11A and 11B Function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

A loss of power to 4160 V Bus 11A and 11B will be accompanied by a loss of power to both MFW pumps and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each bus. Loss of power to both buses will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. Each bus is considered a separate Function for the purpose of this LCO.

f. Auxiliary Feedwater - Trip Of Both Main Feedwater Pumps

A Trip of both MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal. The MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per MFW pump satisfy redundancy requirements with two-out-of-two logic. Each MFW pump is considered a Separate Function for the purpose of this LCO. A trip of both MFW pumps starts both motor driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

(continued)



BASES

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f. Auxiliary Feedwater - Trip Of Both Main Feedwater
Pumps (continued)

This Function must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, 5, and 6 the MFW pumps are not in operation, and thus pump trip is not indicative of a condition requiring automatic AFW initiation.

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. As shown on Figure B 3.3.2-1, the ESFAS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Actuation Logic. Therefore, a channel may be inoperable due to the failure of a field instrument, loss of 120 VAC instrument bus power or a bistable failure which affects one or both ESFAS trains. The only exception to this are the Manual ESFAS and Automatic Actuation Logic Functions which are defined strictly on a train basis. The Automatic Actuation Logic consists of all circuitry housed within the actuation subsystem, including the master relays, slave relays, and initiating relay contacts responsible for activating the ESF equipment.

(continued)



BASES

ACTIONS
(continued)

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one channel or train for one or more Functions are inoperable. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in an ESFAS Function exceed those specified in all related Conditions associated with an ESFAS Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if the ESFAS function is applicable in the current MODE of operation.

B.1

Condition B applies to the AFW-Trip of Both MFW Pumps ESFAS Function. If a channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval.

C.1

If the Required Action and Completion Time of Condition B is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

ACTIONS
(continued)

D.1

Condition D applies to the following ESFAS Functions:

- Manual Initiation of SI;
- Manual Initiation of Steam Line Isolation;
- AFW-SG Water Level-Low Low; and
- AFW-Undervoltage-Bus 11A and 11B.

If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each manual initiation, Function additional AFW actuation channels available besides the SG Water Level-Low Low and Undervoltage-Bus 11A and 11B AFW Initiation Functions, and the low probability of an event occurring during this interval.

E.1

Condition E applies to the automatic actuation logic and actuation relays for the following ESFAS Functions:

- Steam Line Isolation;
- Feedwater Isolation; and
- AFW.

Condition E addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this time interval. The Completion Time of 6 hours is consistent with Reference 7.

(continued)

BASES

ACTIONS
(continued)

F.1

Condition F applies to the following Functions:

- Steam Line Isolation - Containment Pressure - High High;
- Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low;
- Steam Line Isolation - High - High Steam Flow Coincident With Safety Injection; and
- Feedwater Isolation - SG Water Level - High.

Condition F applies to Functions that typically operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. This 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 7.

(continued)



BASES

ACTIONS
(continued)

G.1

If the Required Actions and Completion Times of Conditions D, E, or F are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

Condition H applies to the following ESFAS functions:

- Manual Initiation of CS; and
- Manual Initiation of Containment Isolation.

If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each Function (except for CS) and the low probability of an event occurring during this interval.

I.1

Condition I applies to the automatic actuation logic and actuation relays for the following Functions:

- SI;
- CS; and
- Containment Isolation.

(continued)



BASES

ACTIONS

I.1 (continued)

Condition I addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. The Completion Time of 6 hours is consistent with Reference 7.

J.1

Condition J applies to the following Functions:

- SI-Containment Pressure-High; and
- CS-Containment Pressure-High High.

Condition J applies to Functions that operate on a two-out-of-three logic (for CS-Containment Pressure-High High there are two sets of this logic). Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

(continued)



BASES

ACTIONS
(continued)

K.1

If the Required Actions and Completion Times of Conditions H, I, or J are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

L.1

Condition L applies to the following Functions:

- SI-Pressurizer Pressure-Low; and
- SI-Steam Line Pressure-Low.

Condition L applies to Functions that operate on a two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

(continued)

BASES

ACTIONS
(continued)

M.1

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

N.1

Condition N applies if a AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or SAFW pump must be declared inoperable and the applicable Conditions of LCO 3.7.5, "Auxiliary Feedwater (AFW) System" must be entered immediately. Each AFW manual initiation switch controls one AFW or SAFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

SURVEILLANCE
REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2; Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

Note 1 has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.2.1

This SR is the performance of a CHANNEL CHECK for the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.1 (continued)

CHANNEL CHECK acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

This SR is the performance of a COT every 92 days for the following ESFAS functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.2 (continued)

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found to be within the Allowable Values specified in Table 3.3.2-1 and established plant procedures. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 92 days is consistent with in Reference 7. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

This SR is the performance of a TADOT every 92 days. This test is a check of the AFW-Undervoltage-Bus 11A and 11B Function.

The test includes trip devices that provide actuation signals directly to the protection system. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency of 92 days is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.4

This SR is the performance of a TADOT every 24 months. This test is a check of the SI, CS, Containment Isolation, Steam Line Isolation, and AFW Manual Initiations, and the AFW-Trip of Both MFW Pumps Functions. Each Function is tested up to, and including, the master transfer relay coils. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Manual Initiations, and AFW-Trip of Both MFW Pumps Functions have no associated setpoints.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.2.5

This SR is the performance of a CHANNEL CALIBRATION every 24 months of the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High;
- AFW-SG Water Level-Low Low; and
- AFW-Undervoltage-Bus 11A and 11B.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.2.6

This SR ensures the SI-Pressurizer Pressure-Low and SI-Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig while in MODES 1, 2, and 3. Periodic testing of the pressurizer pressure channels is required to verify the setpoint to be less than or equal to the limit.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 6). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

If the pressurizer pressure interlock setpoint is nonconservative, then the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are considered inoperable. Alternatively, the pressurizer pressure interlock can be placed in the conservative condition (nonbypassed). If placed in the nonbypassed condition, the SR is met and the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions would not be considered inoperable.

SR 3.3.2.7

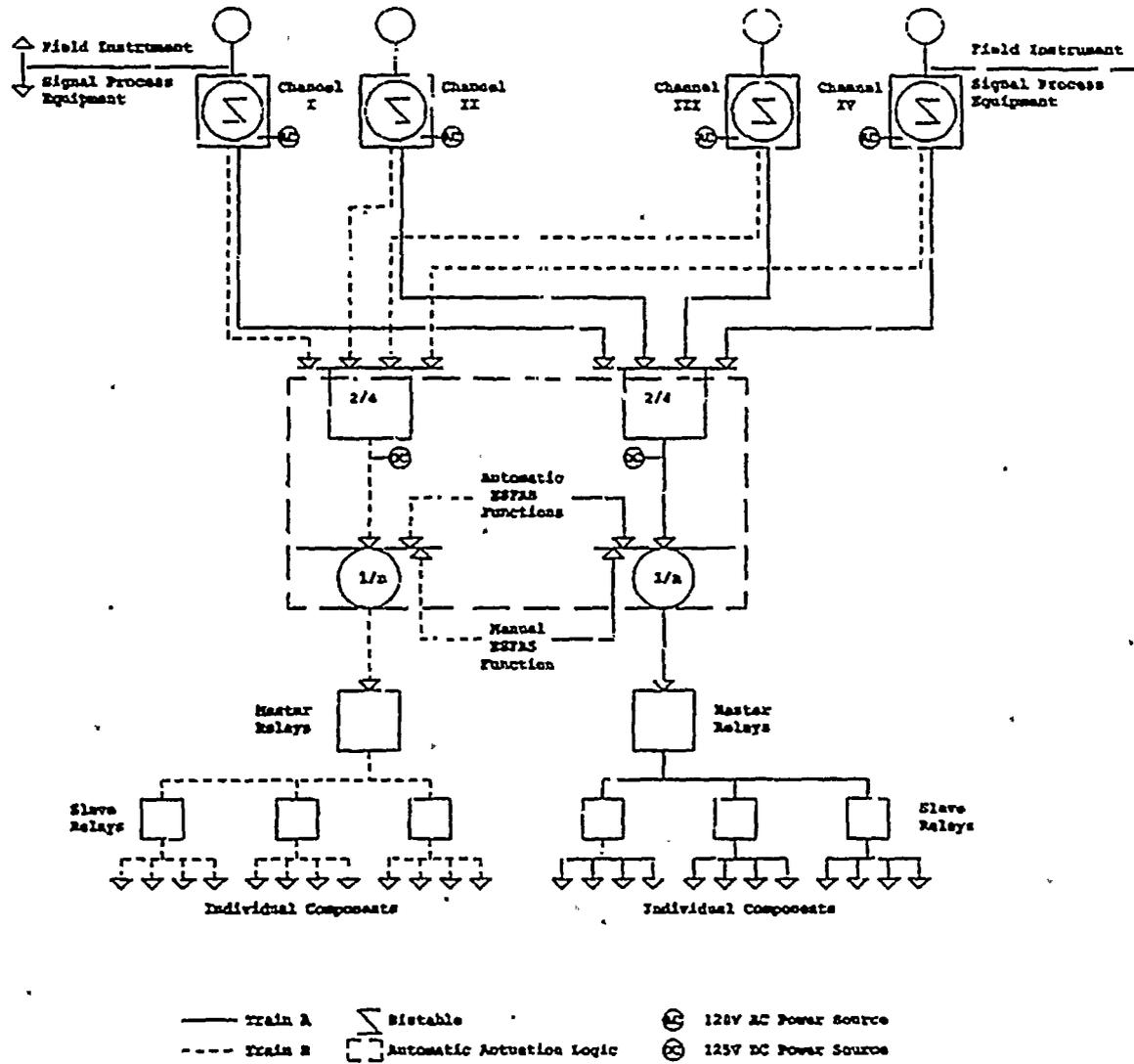
This SR is the performance of an ACTUATION LOGIC TEST on all ESFAS Automatic Actuation Logic and Actuation Relays Functions every 24 months. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock state and verification of the required logic output. Relay and contact operation is verified by a continuance check or actuation of the end device.

The Frequency of 24 months is based on operating experience and the need to perform this testing during a plant shutdown to prevent a reactor trip from occurring.

(continued)

BASES (continued)

- REFERENCES
1. Atomic Industrial Forum (AIF) GDC 15, Issued for Comment July 10, 1967.
 2. UFSAR, Chapter 7.
 3. UFSAR, Chapter 6.
 4. UFSAR, Chapter 15.
 5. IEEE-279-1971.
 6. EWR-5126, "Guidelines For Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
 7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
-



For illustration only

Figure B 3.3.2-1

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident conditions. This instrumentation provides the necessary support for the operator to take required manual actions, verify that automatic and required manual safety functions have been completed, and to determine if fission product barriers have been breached following a Design Basis Accident (DBA).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior during an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified in Reference 1 addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO provide information for key parameters identified during implementation of Regulatory Guide 1.97 as Category I variables. Category I variables are organized into four types and are the key variables deemed risk significant because they are needed to:

- a. Provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs (Type A).
- b. Provide the primary information required for the control room operator to verify that required automatic and manually controlled functions have been accomplished (Type B);

(continued)

BASES

BACKGROUND
(continued)

- c. Provide information to the control room operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release (Type C); and
- d. Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat (Type E).

All Type A and key Type B, C, and E parameters have been identified as Category I variables in Reference I which also provides justification for deviating from the NRC proposed list of Category I variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE
SAFETY ANALYSES

The PAM instrumentation ensures the availability of Regulatory Guide 1.97 Category I variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures for the primary success path of DBAs (e.g., loss of coolant accident (LOCA));
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether required automatic and manual safety functions have been accomplished;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category I, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A, variables are important for reducing public risk and satisfy Criterion 4.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the plant Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident.

This LCO requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from obtaining the information necessary to determine the safety status of the plant, and to bring the plant to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. More than two channels may be required if failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

Table 3.3.3-1 lists all Category I variables identified by Reference 1.

(continued)

BASES

LCO
(continued)

Category I variables are considered OPERABLE when they are capable of providing immediately accessible display and continuous readout in the control room. The Hydrogen Monitors are considered OPERABLE when continuous readout is available in the Control Room or in the relay room. Each channel must also be supplied by separate electrical trains except as noted below. In addition, in accordance with LCO 3.0.6, it is not required to declare a supported system inoperable due to the inoperability of the support system (e.g., electric power). Since the inoperability of Instrument Bus D does not have any associated Required Actions, the loss of this power source may affect the OPERABILITY of the Pressurizer Pressure and SG Water Level (Narrow Range) Functions.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

1. Pressurizer Pressure

Pressurizer Pressure is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer pressure is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of pressure transmitters comprise the two channels required for this function:

- PT-429 and PT-431;
- PT-430 and PT-431;
- PT-429 and PT-449;
- PT-430 and PT-449; or
- PT-431 and PT-449

The loss of Instrument Bus D requires declaring PT-449 inoperable.

(continued)

BASES

LCO
(continued)

2. Pressurizer Level

Pressurizer Level is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer water level is also used to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of level transmitters comprise the two channels required for this function:

- LT-426 and LT-428; or
- LT-427 and LT-428.

3, 4. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures are Category I variables (RCS Cold Leg Temperature is also a Type A variable) provided for verification of core cooling and long term surveillance of RCS integrity.

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify natural circulation in the RCS.

Temperature inputs are provided by two independent temperature sensor resistance elements and associated transmitters in each loop. Temperature elements TE-409B-1 and TE-410B-1 provide the required RCS cold leg temperature input for RCS Loops A and B, respectively. Temperature elements TE-409A-1 and TE-410A-1 provide the required RCS hot leg temperature input for RCS Loops A and B, respectively.

(continued)

BASES

LCO
(continued)

5. RCS Pressure (Wide Range)

RCS wide range pressure is a Type A variable provided for verification of core cooling and the long term surveillance of RCS integrity.

RCS pressure is used to verify delivery of SI flow to the RCS from at least one train when the RCS pressure is below the SI pump shutoff head. RCS pressure is also used to verify closure of manually closed pressurizer spray line valves and pressurizer power operated relief valves (PORVs) and for determining RCS subcooling margin.

RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and stop the residual heat removal pumps (RHR);
- to manually restart the RHR pumps;
- as reactor coolant pump (RCP) trip criteria;
- to make a determination on the nature of the accident in progress and where to go next in the emergency operating procedure; and
- to determine whether to operate the pressurizer heaters.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization.

(continued)

BASES

LCO

5. RCS Pressure (Wide Range) (continued)

RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication.

RCS pressure transmitters PT-420 and PT-420A provide the two required channels for this function.

6. RCS Subcooling Monitor

RCS Subcooling Monitor is a Type A variable provided for verification of core cooling and long term surveillance of RCS integrity. The RCS Subcooling Monitor is used to provide information to the operator, derived from RCS hot leg temperature and RCS pressure, on subcooling. RCS subcooling margin is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

The emergency operating procedures determine RCS subcooling margin based on the core exit thermocouples (CETs) and RCS pressure. Therefore, any of the following combination of parameters comprise the two required channels for this function:

- TI-409A and TI-410A; or
- One pressurizer pressure transmitter and two CETs in each of the four quadrants supplied by electrical train A and train B (i.e., total of two pressurizer pressure transmitters and 16 CETs).

(continued)

BASES

LCO
(continued)

7. Reactor Vessel Water Level

Reactor Vessel Water Level is a Type A variable provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

When both RCPs are stopped, the Reactor Vessel Water Level Indication System (RVLIS) provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. When the RCPs are operating, RVLIS indicates the fluid fraction of the RCS. Measurement of the collapsed water level or fluid fraction is selected because it is a direct indication of the water inventory.

Level transmitters LT-490A and LT-490B provide the two required channels for this function.

8. Containment Sump B Water Level

Containment Sump B Water Level is a Type A variable provided for verification and long term surveillance of RCS integrity.

Containment Sump B Water Level is used to determine:

- containment sump level for accident diagnosis;
- when to begin the recirculation procedure; and
- whether to terminate SI, if still in progress.

Level transmitters LT-942 and LT-943, each with five discrete level switches, provide the two required channels for this function.

(continued)

BASES

LCO
(continued)

9. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is a Type A variable provided for verification of RCS and containment OPERABILITY.

Containment Pressure (Wide Range) is used to determine the type of accident in progress and when, and if, to use emergency operating procedure containment adverse values.

Any of the following combinations of pressure transmitters comprise the two required channels for this function:

- PT-946 and PT-948; or
- PT-950 and PT-948.

10. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) is a Type E Category I variable provided to monitor for the potential of significant radiation releases into containment and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Containment radiation level is used to determine the type of accident in progress (e.g., LOCA), and when, or if, to use emergency operating procedure containment adverse values.

Radiation monitors R-29 and R-30 are used to provide the two required channels for this function.

11. Hydrogen Monitors

Hydrogen Concentration is a Type C Category I variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

(continued)

BASES

LCO

11. Hydrogen Monitors (continued)

Hydrogen monitors HMSLCPA and HMSLCPB provide the two required channels for this function. In addition, the Post Accident Sampling System may take the place of one of these monitors. The PASS system Hydrogen Function is not required to provide continuous readout in the control room or relay room for OPERABILITY.

12. Condensate Storage Tank (CST) Level

CST Level is a Type A variable provided to ensure a water supply is available for the preferred Auxiliary Feedwater (AFW) System. The CST consists of two identical tanks connected by a common outlet header.

CST level is used to determine:

- if sufficient CST inventory is available immediately following a loss of normal feedwater or small break LOCA; and
- when to manually replenish the CST or align the safety related source of water (service water) to the preferred AFW system.

Level transmitters LT-2022A and LT-2022B provide the two required channels for this function. However, only the level transmitter associated with the CST(s) required by LCO 3.7.6, "Condensate Storage Tank(s)" are required for this LCO.

13. Refueling Water Storage Tank (RWST) Level

RWST Level is a Type A variable provided for verifying a water source to the SI, RHR, and Containment Spray (CS) Systems.

(continued)

BASES

LCO

13. Refueling Water Storage Tank (RWST) Level (continued)

The RWST level accuracy is established to allow an adequate supply of water to the SI, RHR, and CS pumps during the switchover to the recirculation phase of an accident. A high degree of accuracy is required to maximize the time available to the operator to complete the switchover to the sump recirculation phase and ensure sufficient water is available to maintain adequate net positive suction head (NPSH) to operating pumps..

Level transmitters LT-920 and LT-921 provide the two required channels for this function.

14. RHR Flow

RHR Flow is a Type A variable provided for verifying low pressure safety injection to the reactor vessel and to the CS and SI pumps.

RHR flow is used to determine when to stop the RHR pumps and if sufficient flow is available to the CS and SI pumps during recirculation.

Since different flow transmitters are used to verify injection to the reactor vessel and to verify flow to the CS and SI pumps, FT-626 and FT-931A comprise one required channel and FT-689 and FT-931B comprise a second required channel.

(continued)



BASES

LCO

15, 16, 17, 18. Core Exit Temperature
(continued)

Core Exit Temperature is a Type A variable provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid CETs necessary for measuring core cooling. The evaluation determined the necessary complement of CETs required to detect initial core recovery and trend the ensuing core heatup. The evaluation accounted for core nonuniformities, including incore effects of the radial decay power distribution, excore effects of reflux in the hot legs, and nonuniform inlet temperatures. Based on these evaluations, adequate core cooling is ensured with two valid Core Exit Temperature channels per quadrant with two CETs per required channel.

Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for plant stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Because of the small core size, two randomly selected thermocouples are sufficient to meet the two thermocouples per channel requirement in any quadrant. However, a CET which lies directly on the dividing line between two quadrants can only be used to satisfy the minimum required channels for one quadrant.

A CET is considered OPERABLE when it is within $\pm 35^{\circ}\text{F}$ of the average CET reading except for the CETs associated with peripheral assemblies. These CETs (A7, B5, C3, C.11, D2, D12, H13, I2, K3, K11, L10, and M6) are considered OPERABLE when they are within $\pm 43^{\circ}\text{F}$ of the average CET reading. At least two CETs from each of the following trains must be OPERABLE in each of the four quadrants:

(continued)

BASES

LCO

15, 16, 17, 18. Core Exit Temperature (continued)

Train A		Train B	
<u>CET</u>	<u>Location</u>	<u>CET</u>	<u>Location</u>
T2	M6	T1	I4
T5	J3	T3	L7
T6	I2	T4	K3
T7	J6	T10	J9
T8	L10	T13	K11
T9	J8	T14	D12
T12	H6	T16	H10
T15	H9	T17	E10
T18	F8	T19	G7
T21	C11	T20	C8
T22	H11	T24	F12
T23	H13	T25	G12
T26*	I10	T27	E6
T28	D5	T29	E4
T33	D2	T30*	G4
T34	C3	T31	G2
T36	B7	T32*	G1
T38	B5	T35	A7
T39	D7	T37	C6

* - These thermocouples are in the reactor vessel head and cannot be credited with respect to this LCO.

19, 20. AFW Flow

AFW Flow is a Type A variable provided to monitor operation of the preferred AFW system.

(continued)

BASES

LCO 19, 20. AFW Flow (continued)

The AFW System provides decay heat removal via the SGs and is comprised of the preferred AFW System and the Standby AFW (SAFW) System. The use of the preferred AFW or SAFW System to provide this decay heat removal function is dependent upon the type of accident. AFW flow indication is required from the three pump trains which comprise the preferred AFW System since these pumps automatically start on various actuation signals. The failure of the preferred AFW System (e.g., due to a high energy line break (HELB) in the Intermediate Building) is detected by AFW flow indication. At this point, the SAFW System is manually aligned to provide the decay heat removal function.

SAFW flow can also be used to verify that AFW flow is being delivered to the SGs. However, the primary indication of this is provided by SG water level. Therefore, flow indication from the SAFW pumps is not required.

Each of the three preferred AFW pump trains has two redundant transmitters; however, only the flow transmitter supplied power from the same electrical train as the AFW pump is required for this LCO. Therefore, flow transmitters FT-2001 (MCB indicator FI-2021A) and FT-2007 (MCB indicator FI-2024A) comprise the two required channels for SG A and FT-2002 (MCB indicator FI-2022A) and FT-2006 (MCB indicator FI-2023A) comprise the two required channels for SG B.

(continued)

BASES

LCO

21, 22, 23, 24.
(continued)

SG Water Level (Narrow and Wide Range)

SG Water Level is a Type A variable provided to monitor operation of decay heat removal via the SGs. For the narrow range level, the signals from the transmitters are independently indicated on the main control board as 0% to 100%. This corresponds to approximately above the top of the tube bundles to the top of the swirl vane separators (span of 143 inches). For the wide range level, signals from the transmitters are indicated as 0 to 520 inches (0% to 100%) on the main control board.

SG Water Level (Narrow and Wide Range) is used to:

- identify the faulted SG following a tube rupture;
- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify an SGTR); and
- verify plant conditions for termination of SI during secondary plant HELBs outside containment.

Redundant monitoring capability is provided by two trains of instrumentation per SG.

SG Water Level (Narrow Range) requires 2 channels of indication per SG. This can be met using any of the following combinations of level transmitters for SG A:

- LT-461 and LT-462;
- LT-462 and LT-463; or
- LT-461 and LT-463;

(continued)

BASES

LCO

21, 22, 23, 24. SG Water Level (Narrow and Wide Range) (continued)

For SG B, any of the following combinations of level transmitters can be used:

- LT-471 and LT-473;
- LT-471 and LT-472; or
- LT-472 and LT-473.

The loss of Instrument Bus D requires declaring LT-463 and LT-471 inoperable.

SG Water Level (Wide Range) requires 2 channels of indication per SG. Two channels per SG are required since the loss of one channel with no backup available may result in the complete loss of information required by the operators to accomplish necessary safety functions. Level transmitters LT-504 and LT-505 comprise the two required channels for SG A and LT-506 and LT-507 comprise the two required channels for SG B.

25, 26. SG Pressure

SG Pressure is a Type A variable provided to monitor operation of decay heat removal via the SGs. The signals from the transmitters are calibrated for a range of 0 psig to 1400 psig. Redundant monitoring capability is provided by three available trains of instrumentation.

Any of the following combinations of pressure transmitters comprise the two required channels for SG A:

- PT-468 and PT-482; or
- PT-469 and PT-482.

(continued)

BASES

LCO 25 and 26. SG Pressure (continued)

Any of the following combinations of pressure transmitters comprise the two required channels for SG B:

- PT-479 and PT-478; or
 - PT-478 and PT-483.
-

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, the PAM instrumentation is not required to be OPERABLE because plant conditions are such that the likelihood of an event that would require PAM instrumentation is low.

ACTIONS

The ACTIONS are modified by two Notes.

Note 1 has been added to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies when one or more Functions have one required channel that is inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM Instrumentation during this interval.

Condition A is modified by a Note which states that the Condition is not applicable to Table 3.3.3-1 Functions 3 and 4. These Functions are addressed by Condition C which provides the necessary required actions for these single channel Functions.

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A is not met. This Condition requires the immediate initiation of actions to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the Condition is entered. This report shall discuss the results of the root cause evaluation of the inoperability and identify proposed restorative actions or alternate means of providing the required function. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation. If alternate means are to be used, they must be developed and tested prior to submittal of the special report.

(continued)

BASES

ACTIONS
(continued)

C.1

Condition C applies when a Function has one inoperable required channel and no diverse channel OPERABLE (i.e., loss of RCS Hot Leg Temperature or RCS Cold Leg Temperature Functions). This Condition requires restoring the inoperable channel in the affected Function to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with a complete loss of function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of the inoperable channel limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Condition C is modified by a Note which states that this Condition is only applicable to Table 3.3.3-1 Functions 3 and 4. All remaining Functions are addressed by Condition A with one channel inoperable.

D.1

Condition D applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action D.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. Condition D is modified by a Note that excludes Function 11 since the inoperability of two hydrogen monitor channels is addressed by Condition E.

(continued)

BASES

ACTIONS
(continued)

E.1

Condition E applies when two hydrogen monitor channels are inoperable. This Condition requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA which would potentially require use of the hydrogen recombiners would occur during this time.

F.1

Condition F applies when the Required Action and associated Completion Time of Condition C, D, or E are not met. Required Action F.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C, D, or E, and the associated Completion Time has expired, Condition F is entered for that channel and provides for transfer to the appropriate subsequent Condition.

G.1 and G.2

If one channel for Function 3 and 4 cannot be restored to OPERABLE status within the required Completion Time for Condition C, if one channel for Function 1, 2, 3, 4, 5, 6, 8, 9, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, or 26 cannot be restored to OPERABLE status within the required Completion Time of Condition D, or if one channel for Function 11 cannot be restored to OPERABLE status within the required Completion Time of Condition E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

H.1

If one channel for Function 7 or 10 cannot be restored to OPERABLE status within the required Completion Time of Condition D, the plant must take immediate action to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the action is required. This report discusses the alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation, the degree to which the alternate means are equivalent to the installed PAM channels, the areas in which they are not equivalent, and a schedule for restoring the normal PAM channels.

These alternate means must have been developed and tested and may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

(continued)

BASES

SURVEILLANCE

SR 3.3.3.1 (continued)

Channel check acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors shall include an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. UFSAR, Section 7.5.2.
 2. Regulatory Guide 1.97, Rev. 3.
 3. NUREG-0737, Supplement 1, "TMI Action Items."
-

B 3.3 INSTRUMENTATION

B 3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe plant operation. The LOP DG start instrumentation consists of two channels on each of safeguards Buses 14, 16, 17, and 18 (Ref. 1). Each channel contains one loss of voltage relay and one degraded voltage relay (see Figure B 3.3.4-1). A one-out-of-two logic in both channels will cause the following actions on the associated safeguards bus:

- a. trip of the normal feed breaker from offsite power;
- b. trip of the bus-tie breaker to the opposite electrical train (if closed);
- c. shed of all bus loads except the CS pump, component cooling water pump (if no safety injection signal is present), and safety related motor control centers; and
- d. start of the associated DG.

The degraded voltage logic is provided on each 480 V safeguards bus to protect Engineered Safety Features (ESF) components from exposure to long periods of reduced voltage conditions which can result in degraded performance and to ensure that required motors can start. The loss of voltage logic is provided on each 480 V safeguards bus to ensure the DG is started within the time limits assumed in the accident analysis to provide the required electrical power if offsite power is lost.

The degraded voltage relays have time delays which have inverse operating characteristics such that the lower the bus voltage, the faster the operating time. The loss of voltage relays have definite time delays which are not related to the rate of the loss of bus voltage. These time delays are set to permit voltage transients during worst case motor starting conditions.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The LOP DG start instrumentation is required for the ESF Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS). Undervoltage conditions which occur independent of any accident conditions result in the start and bus connection of the associated DG, but no automatic loading occurs.

Accident analyses credit the loading of the DG based on the loss of offsite power during a Design Basis Accident (DBA). The most limiting DBA of concern is the large break loss of coolant accident (LOCA) which requires ESF Systems in order to maintain containment integrity and protect fuel contained within the reactor vessel (Ref. 2). The detection and processing of an undervoltage condition, and subsequent DG loading, has been included in the delay time assumed for each ESF component requiring DG supplied power following a DBA and loss of offsite power.

The loss of offsite power has been assumed to occur either coincident with the DBA or at a later period (40 to 90 seconds following the reactor trip) due to a grid disturbance caused by the turbine generator trip. If the loss of offsite power occurs at the same time as the safety injection (SI) signal parameters are reached, the accident analyses assumes the SI signal will actuate the DG within 2 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12 seconds total time). If the loss of offsite power occurs before the SI signal parameters are reached, the accident analyses assumes the LOP DG start instrumentation will actuate the DG within 2.75 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12.75 seconds total time). If the loss of offsite power occurs after the SI signal parameters are reached (grid disturbance), the accident analyses assumes the LOP DG start instrumentation will open the feeder breaker to the affected bus within 2.75 seconds and the DG will connect to the bus within an additional 1.5 seconds (DG was actuated by SI signal). The grid disturbance has been evaluated based on a 140°F peak clad temperature penalty during a LOCA and demonstrated to result in acceptable consequences.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The degraded voltage and undervoltage setpoints are based on the minimum voltage required for continued operation of ESF Systems assuming worst case loading conditions (i.e., maximum loading upon DG sequencing). The Trip Setpoint for the loss of voltage relays, and associated time delays, have been chosen based on the following considerations:

- a. Actuate the associated DG within 2.75 seconds as assumed in the accident analysis; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest assumed voltage drop below the voltage setting and the reset value of the trip function.

The Trip Setpoint for the degraded voltage channels, and associated time delays, have been chosen based on the following considerations;

- a. Prevent motors supplied by the 480 V bus from operating at reduced voltage conditions for long periods of time; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available, and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest voltage drop below the maximum voltage setting and the reset value of the trip function.

The LOP DG start instrumentation channels satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

This LCO requires that each 480 V safeguards bus have two OPERABLE channels of the LOP DG start instrumentation in MODES 1, 2, 3, and 4 when the associated DG supports safety systems associated with the ESFAS. In MODES 5 and 6, the LOP DG start instrumentation channels for each 480 V safeguards bus must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents.

The LOP start instrumentation is considered OPERABLE when two channels, each comprised of one degraded voltage and one loss of voltage relays are available for each 480 V safeguards bus (i.e., Bus 14, 16, 17, and 18). Each of the LOP channels must be capable of detecting undervoltage conditions within the voltage limits and time delays assumed in the accident analysis.

The Allowable Values and Trip Setpoints for the degraded voltage and loss of voltage Functions are specified in SR 3.3.4.2. The Allowable Values specified in SR 3.3.4.2 are those setpoints which ensure that the associated DG will actuate within 2.75 seconds on undervoltage conditions, and that the DG will not actuate on momentary voltage drops which could affect ESF actuation times as assumed in the accident analysis. The Trip Setpoints specified in SR 3.3.4.2 are the nominal setpoints selected to ensure that the setpoint measured by the Surveillance does not exceed the Allowable Value accounting for maximum instrument uncertainties between scheduled surveillances. Therefore, LOP start instrumentation channels are OPERABLE when the CHANNEL CALIBRATION "as left" value is within the Trip Setpoint limits and the CHANNEL CALIBRATION and TADOT "as found" value is within the Allowed Value setpoints. The basis for all setpoints is contained in Reference 3.

(continued)

BASES (continued)

APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the 480 V safeguards buses.

ACTIONS In the event a relay's Trip Setpoint is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then the channel must be declared inoperable and the LCO Condition entered as applicable.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. This Note states that separate Condition entry is allowed for each 480 V safeguards bus.

A.1

With one or more 480 V bus(es) with one channel inoperable, Required Action A.1 requires the inoperable channel(s) to be placed in trip within 6 hours. With an undervoltage channel in the tripped condition, the LOP DG start instrumentation channels are configured to provide a one-out-of-one logic to initiate a trip of the incoming offsite power for the respective bus. The remaining OPERABLE channel is comprised of one-out-of-two logic from the degraded and loss of voltage relays. Any additional failure of either of these two OPERABLE relays requires entry into Condition B.

(continued)

BASES

ACTIONS
(continued)

B.1

Condition B applies to the LOP DG start Function when the Required Action and associated Completion Time for Condition A are not met or with one or more 480 V bus(es) with two channels of LOP start instrumentation inoperable.

Condition B requires immediate entry into the Applicable Conditions specified in LCO 3.8.1, "AC Sources—MODES 1, 2, 3, and 4," or LCO 3.8.2, "AC Sources—MODES 5 and 6," for the DG made inoperable by failure of the LOP DG start instrumentation. The actions of those LCOs provide for adequate compensatory actions to assure plant safety.

SURVEILLANCE
REQUIREMENTS

The Surveillances are modified by a Note to indicate that, when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 4 hours, provided the second channel maintains trip capability. Upon completion of the Surveillance, or expiration of the 4 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the assumption that 4 hours is the average time required to perform channel surveillance. Based on engineering judgement, the 4 hour testing allowance does not significantly reduce the probability that the LOP DG start instrumentation will trip when necessary.

SR 3.3.4.1

This SR is the performance of a TADOT every 31 days. This test checks trip devices that provide actuation signals directly. For these tests, the relay Trip Setpoints are verified and adjusted as necessary to ensure Allowable Values can still be met. The 31 day Frequency is based on the known reliability of the relays and controls and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.4.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling of the LOP DG start instrumentation for each 480 V bus.

The voltage setpoint verification, as well as the time response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience consistent with the typical industry refueling cycle and is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 8.3.
 2. UFSAR, Chapter 15.
 3. RG&E Design Analysis DA-EE-93-006-08, "480 Volt Undervoltage Relay Settings and Test Acceptance Criteria."
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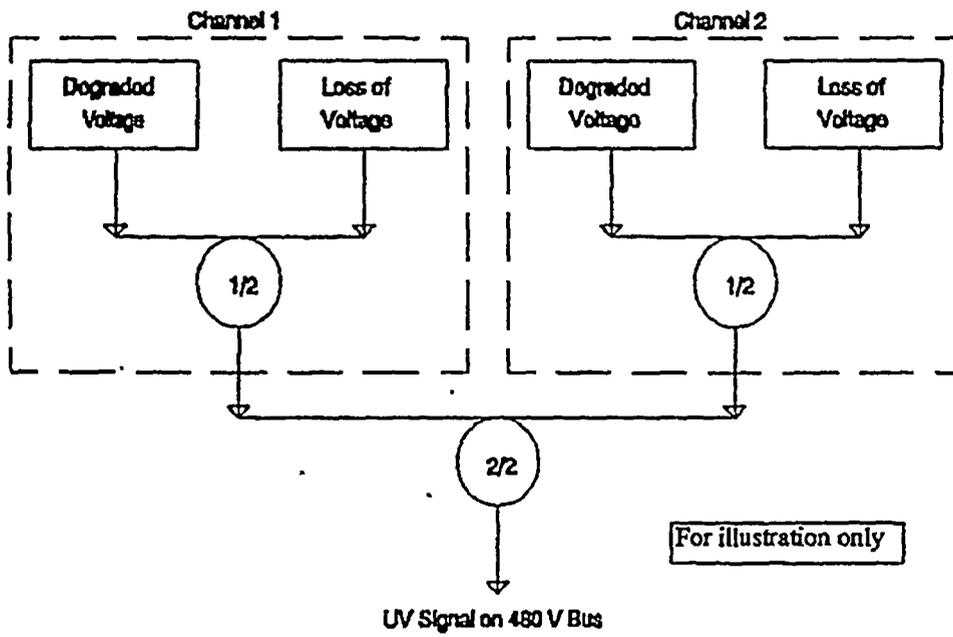


Figure B 3.3.4-1
DG LOP Instrumentation

B 3.3 INSTRUMENTATION

B 3.3.5 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation instrumentation closes the containment isolation valves in the Mini-Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-Purge System may be used in all MODES while the Shutdown Purge System may only be used with the reactor shutdown.

Containment ventilation isolation initiates on a containment isolation signal, containment radiation signal, or by manual actuation of containment spray (CS). The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss the containment isolation and manual containment spray modes of initiation.

Two containment radiation monitoring channels are provided as input to the containment ventilation isolation. The two radiation detectors are of different types: gaseous (R-12), and particulate (R-11). Both detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the two channels are not considered redundant. Instead, they are treated as two one-out-of-one Functions. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

(continued)

BASES

BACKGROUND
(continued)

The Mini-Purge System has inner and outer containment isolation valves in its supply and exhaust ducts while the Shutdown-Purge System only has one valve located outside containment since the inside valve was replaced by a blind flange that is used during MODES 1, 2, 3, and 4. A high radiation signal from any one of the two channels initiates containment ventilation isolation, which closes all isolation valves in the Mini-Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABLE
SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for accident mitigation functions isolated early in the event, within approximately 60 seconds. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the containment isolation signal to ensure closing of the ventilation valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown even though containment isolation is not specifically credited for this event. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accident offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.5-1, is OPERABLE.

(continued)

BASES

LCO

1. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 2.b, Containment Spray-Manual Initiation, and ESFAS Function 3, Containment Isolation. The applicable MODES and specified conditions for the containment ventilation isolation portion of these Functions are different and less restrictive than those for their respective CS and ESFAS roles. If one or more of the CS or containment isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their respective isolation Functions in LCO 3.3.2 need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

2. Containment Radiation

The LCO specifies two required channels of radiation monitors (R-11 and R-12) to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur.

3. Containment Isolation

Refer to LCO 3.3.2, Function 3, for all initiating Functions and requirements.

(continued)

BASES

LCO
(continued)

4. Containment Spray - Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays, Containment Isolation, Containment Spray - Manual Initiation, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

(continued)

BASES

ACTIONS
(continued)

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Since the two containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hour allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

Condition C applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place each valve in its closed position or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.1 (continued)

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.5.2

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. The setpoint shall be left consistent with the current plant specific calibration procedure tolerance.

SR 3.3.5.3

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 24 months. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.5.4

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.
 2. NUREG-1366.
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B 3.3 INSTRUMENTATION

B 3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation
Instrumentation

BASES

BACKGROUND

The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This system is described in the Bases for LCO 3.7.9, "Control Room Emergency Air Treatment System (CREATS)." This LCO only addresses the actuation instrumentation for the high radiation state CREATS Mode F.

The high radiation state CREATS Mode F actuation instrumentation consists of noble gas (R-36), particulate (R-37), and iodine (R-38) radiation monitors. These detectors are located on the operating level on the Turbine Building and utilize a common air supply pump. A high radiation signal from any of these detectors will initiate the CREATS filtration train and isolate each air supply path with two dampers. The control room operator can also initiate the CREATS filtration train and isolate the air supply paths by using a manual pushbutton in the control room.

APPLICABLE
SAFETY ANALYSES

The location of components and CREATS related ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 1). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

In MODES 5 and 6, and during movement of irradiated fuel assemblies, the CREATS ensures control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident.

The CREATS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREATS is OPERABLE.

1. Manual Initiation

The LCO requires one train to be OPERABLE. The train consists of one pushbutton and the interconnecting wiring to the actuation logic. The operator can initiate the CREATS Filtration train at any time by using a pushbutton in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals required by this LCO.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires one train of Actuation Logic and Actuation Relays to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation system, including the initiation relay contacts responsible for actuating the CREATS.

3. Control Room Radiation Intake Monitor

The LCO specifies single channels of iodine (R-38), noble gas (R-36), and particulate (R-37) of the Control Room Intake Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREATS filtration train and isolation dampers remains OPERABLE.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS actuation instrumentation must be OPERABLE to control operator exposure during and following a Design Basis Accident.

In MODE 5 or 6, the CREATS actuation instrumentation is required to cope with the release from the rupture of a waste gas decay tank.

During movement of irradiated fuel assemblies, the CREATS actuation instrumentation must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

The most common cause of channel inoperability is failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. The "as left" Trip Setpoint must be within the tolerance specified by the calibration procedure. If the "as found" Trip Setpoint exceeds the limits specified in Table 3.3.5-1, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel/train of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies to one or more Functions with one channel of the CREATS actuation instrumentation inoperable.

If one or more radiation monitor channels, the manual initiation train, or the automatic actuation logic train is inoperable, action must be taken to restore OPERABLE status within 1 hour or isolate the control room from outside air. In this Condition for the manual initiation train inoperable or a radiation monitor channel inoperable, the remaining CREATS actuation instrumentation is adequate to perform the control room protection function but the actuation time or responsiveness of the CREATS may be affected. In this Condition for the automatic actuation logic train inoperable or all radiation monitor channels inoperable, the CREATS is not capable of performing its intended automatic function. This is considered a loss of safety function. The CREATS, however, may still be capable of being placed in CREATS Mode F by manual operator actions. The 1 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to automatically isolate the control room or be manually isolated by the operator.

The Required Action for Condition A is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

Condition B applies when the Required Action and associated Completion Time of Condition A has not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE that minimizes accident risk. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C.3

Condition C applies when the Required Action and associated Completion Time of Condition A has not been met in MODE 5, or 6, or during movement of irradiated fuel assemblies. Actions must be initiated immediately to restore the inoperable channel(s) to OPERABLE status to ensure adequate isolation capability in the event of a waste gas decay tank rupture. Movement of irradiated fuel assemblies and CORE ALTERATIONS must also be suspended immediately to reduce the risk of accidents that would require CREATS actuation. This places the plant in a condition that minimizes risk. This does not preclude movement of fuel or other components to a safe position.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which CREATS Actuation Functions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.1

This SR is the performance of a COT once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.6.2

This SR is the performance of a TADOT of the Manual Actuation Functions every 24 months. The Manual Actuation Function is tested up to, and including, the master relay coils.

The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints associated with them.

SR 3.3.6.3

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.

(continued)

BASES

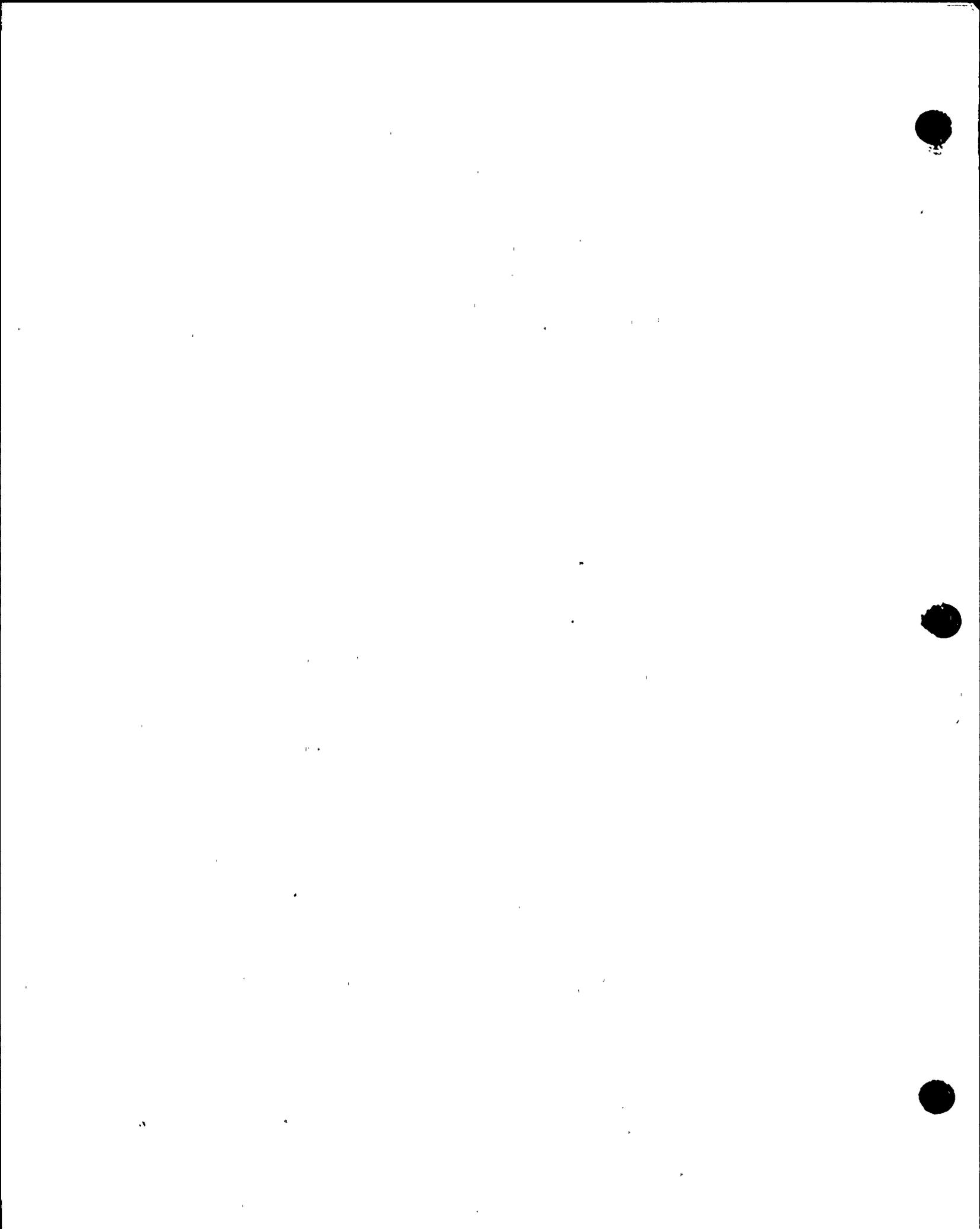
SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.4

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations are tested for the CREATS actuation instrumentation. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is acceptable based on instrument reliability and operating experience.

REFERENCES

1. UFSAR, Section 6.4.
-



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

-----NOTE-----
Pressurizer pressure limit does not apply during pressure transients due to:

- a. THERMAL POWER ramp > 5% RTP per minute; or
 - b. THERMAL POWER step > 10% RTP.
-

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is within limit specified in the COLR.	12 hours
SR 3.4.1.2 Verify RCS average temperature is within limit specified in the COLR.	12 hours
SR 3.4.1.3 -----NOTE----- Required to be performed within 7 days after \geq 95% RTP. ----- Verify RCS total flow rate is within the limit specified in the COLR.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be $\geq 540^\circ\text{F}$.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or both RCS loops not within limit.	A.1 Be in MODE 2 with $K_{eff} < 1.0$.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop $\geq 540^\circ\text{F}$.	Within 30 minutes prior to achieving criticality.
SR 3.4.2.2 -----NOTE----- Only required if any RCS loop $T_{avg} < 547^\circ\text{F}$ and the low T_{avg} alarm is either inoperable or not reset. ----- Verify RCS T_{avg} in each loop $\geq 540^\circ\text{F}$.	Once within 30 minutes and every 30 minutes thereafter

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Action A.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in MODE 1, 2, 3, or 4.</p>	<p>A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.</p>	<p>30 minutes 72 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < 500 psig.</p>	<p>6 hours 36 hours</p>

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODE 1 > 8.5% RTP

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODE 1 > 8.5% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 1 \leq 8.5% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1. Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODES 1 ≤ 8.5% RTP, 2, and 3

LCO 3.4.5 Two RCS loops shall be OPERABLE and one loop shall be in operation.

-----NOTE-----
Both reactor coolant pumps may be de-energized in MODE 3 for ≤ 1 hour per 8 hour period provided:

- a. No operations are permitted that would cause reduction of the RCS boron concentration; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODES 1 ≤ 8.5% RTP,
MODES 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Verify SDM is within limits specified in the COLR. <u>AND</u> A.2 Restore inoperable RCS loop to OPERABLE status.	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours
C. Both RCS loops inoperable. <u>OR</u> No RCS loop in operation.	C.1 De-energize all CRDMs.	Immediately
	<u>AND</u>	
	C.2 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u>	
	C.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1 Verify required RCS loop is in operation.	12 hours
SR 3.4.5.2 Verify steam generator secondary side water levels are \geq 16% for two RCS loops.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.5.3 Verify correct breaker alignment and indicated power are available to the required RCP that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and RHR pumps may be de-energized for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. No RCP shall be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless:
 - a. The secondary side water temperature of each steam generator (SG) is $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures; or
 - b. The pressurizer water volume is < 324 cubic feet (38% level).

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One RCS loop inoperable.</p> <p><u>AND</u></p> <p>Two RHR loops inoperable.</p>	<p>A.1 Initiate action to restore a second loop to OPERABLE status.</p>	<p>Immediately</p>
<p>B. One RHR loop inoperable.</p> <p><u>AND</u></p> <p>Two RCS loops inoperable.</p>	<p>-----NOTE----- Required Action B.1 is not applicable if all RCS and RHR loops are inoperable and Condition C is entered. -----</p> <p>B.1 Be in MODE 5.</p>	<p>24 hours</p>
<p>C. All RCS and RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RCS or RHR loop in operation.</p>	<p>C.1 Suspend all operations involving a reduction of RCS boron concentration.</p> <p><u>AND</u></p> <p>C.2 Initiate action to restore one loop to OPERABLE status and operation.</p>	<p>Immediately</p> <p>Immediately</p>



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2 Verify SG secondary side water level is $\geq 16\%$ for each required RCS loop.	12 hours
SR 3.4.6.3 Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

- a. One additional RHR loop shall be OPERABLE; or
- b. The secondary side water level of at least one steam generator (SG) shall be $\geq 16\%$.

-----NOTES-----

1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless:
 - a. The secondary side water temperature of each SG is $\leq 50^\circ\text{F}$ above each of the RCS cold leg temperatures; or
 - b. The pressurizer water volume is < 324 cubic feet (38% level).
4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. <u>AND</u> Both SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately
B. Both RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2 Verify SG secondary side water level is $\geq 16\%$ in the required SG.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.7.3 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

- NOTES-----
1. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided:
 - a. No operations are permitted that would cause a reduction of the RCS boron concentration;
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature; and
 - c. No draining operations to further reduce the RCS water volume are permitted.
 2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
-

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Both RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving reduction in RCS boron concentration.	Immediately
	<u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2 Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours
	<u>AND</u>	
	A.2 Be in MODE 4.	12 hours
B. Pressurizer heaters capacity not within limits.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is \leq 87%.	12 hours
SR 3.4.9.2 Verify total capacity of the pressurizer heaters is \geq 100 Kw.	92 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2410 psig and ≤ 2544 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures greater than the LTOP enable temperature specified in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Both pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.10.1 -----NOTE----- Required to be performed within 36 hours of entering MODE 4 from MODE 5 with all RCS cold leg temperatures greater than the LTOP enable temperature specified in the PTLR for the purpose of setting the pressurizer safety valves under ambient (hot) conditions only provided a preliminary cold setting was made prior to heatup. ----- Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 1\%$.</p>	<p>In accordance with the Inservice Testing Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. Separate entry into Condition A is allowed for each PORV.
 2. Separate entry into Condition C is allowed for each block valve.
 3. LCO 3.0.4 is not applicable.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both PORVs OPERABLE and not capable of being automatically controlled.	A.1 Close and maintain power to associated block valve.	1 hour
	<u>OR</u>	
	A.2 Place associated PORV in manual control.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One PORV inoperable.	B.1 Close associated block valve.	1 hour
	<u>AND</u>	
	B.2 Remove power from associated block valve.	1 hour
	<u>AND</u>	
	B.3 Restore PORV to OPERABLE status.	72 hours
C. One block valve inoperable.	C.1 Place associated PORV in manual control.	1 hour
	<u>AND</u>	
	C.2 Restore block valve to OPERABLE status.	7 days
D. Both block valves inoperable.	D.1 Place associated PORVs in manual control.	1 hour
	<u>AND</u>	
	D.2 Restore at least one block valve to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. <u>AND</u>	6 hours
	E.2 Be in MODE 4.	12 hours
F. Two PORVs inoperable.	F.1 Initiate action to restore one PORV to OPERABLE status. <u>AND</u>	Immediately
	F.2 Close associated block valves. <u>AND</u>	1 hour
	F.3 Remove power from associated block valves. <u>AND</u>	1 hour
	F.4 Be in MODE 3 with $T_{avg} < 500^{\circ}F$.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.11.1 -----NOTE----- Not required to be performed with block valve closed per LCO 3.4.13. ----- Perform a complete cycle of each block valve.	92 days
SR 3.4.11.2 Perform a complete cycle of each PORV.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with the Emergency Core Cooling System (ECCS) accumulators isolated and either a or b below.

- a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR and no safety injection (SI) pump capable of injecting into the RCS.
- b. The RCS depressurized and an RCS vent of ≥ 1.1 square inches and a maximum of one SI pump capable of injecting into the RCS.

-----NOTES-----

1. The PORVs and an RCS vent ≥ 1.1 square inches are not required to be OPERABLE during performance of the secondary side hydrostatic tests. However, no SI pump may be capable of injecting into the RCS during this test.
 2. ECCS accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.
-

APPLICABILITY: MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or when the RHR system is in the RHR mode of operation,
MODE 5 when the SG primary system manway and pressurizer manway are closed and secured in position,
MODE 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Only applicable to LCO 3.4.12.a. -----</p> <p>One or more SI pumps capable of injecting into the RCS.</p>	<p>A.1 Initiate action to verify no SI pump is capable of injecting into the RCS.</p>	<p>Immediately</p>
<p>B. -----NOTE----- Only applicable to LCO 3.4.12.a. -----</p> <p>One required PORV inoperable in MODE 4.</p>	<p>B.1 Restore required PORV to OPERABLE status.</p>	<p>7 days</p>
<p>C. -----NOTE----- Only applicable to LCO 3.4.12.a. -----</p> <p>One required PORV inoperable in MODE 5 or MODE 6.</p>	<p>C.1 Restore required PORV to OPERABLE status.</p>	<p>72 hours</p>
<p>D. -----NOTE----- Only applicable to LCO 3.4.12.b. -----</p> <p>Two or more SI pumps capable of injecting into the RCS.</p>	<p>D.1 Initiate action to verify a maximum of one SI pump is capable of injecting into the RCS.</p>	<p>Immediately</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. An ECCS accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR.</p>	<p>E.1 Isolate affected ECCS accumulator.</p>	<p>1 hour</p>
<p>F. Required Action and associated Completion Time of Condition E not met.</p>	<p>F.1 Increase RCS cold leg temperature to greater than the LTOP enable temperature specified in the PTLR.</p> <p><u>OR</u></p> <p>F.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.</p>	<p>12 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Two required PORVs inoperable for LCO 3.4.12.a.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A, B, C, or F not met.</p> <p><u>OR</u></p> <p>LTOP System inoperable for any reason other than Condition A, B, C, or E.</p>	<p>G.1 Verify at least one charging pump is in the pull-stop position.</p>	1 hour
	<p><u>AND</u></p> <p>G.2 Depressurize RCS and establish RCS vent of ≥ 1.1 square inches.</p>	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.1 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.a. ----- Verify no SI pump is capable of injecting into the RCS.</p>	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.2 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. -----</p> <p>Verify a maximum of one SI pump is capable of injecting into the RCS.</p>	<p>12 hours</p>
<p>SR 3.4.12.3 -----NOTE----- Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. -----</p> <p>Verify each ECCS accumulator motor operated isolation valve is closed.</p>	<p>Once within 12 hours and every 12 hours thereafter</p>
<p>SR 3.4.12.4 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. -----</p> <p>Verify RCS vent \geq 1.1 square inches open.</p>	<p>12 hours for unlocked open vent valve(s)</p> <p><u>AND</u></p> <p>31 days for locked open vent valve(s)</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	72 hours
SR 3.4.12.6	<p>-----NOTE----- Required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. -----</p> <p>Perform a COT on each required PORV, excluding actuation.</p>	31 days
SR 3.4.12.7	<p>-----NOTE----- Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. -----</p> <p>Verify power is removed from each ECCS accumulator motor operated isolation valve operator.</p>	Once within 12 hours and every 31 days thereafter
SR 3.4.12.8	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE; and
- d. 0.1 gpm total primary to secondary LEAKAGE through each steam generator (SG) when averaged over 24 hours.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. <u>OR</u> RCS pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1 -----NOTE----- Only required to be performed during steady state operation. ----- Perform RCS water inventory balance.</p>	<p>Once during initial 12 hours of steady state operation <u>AND</u> 72 hours thereafter</p>
<p>SR 3.4.13.2 Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.</p>	<p>In accordance with the Steam Generator Tube Surveillance Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flowpaths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 or SR 3.4.14.2 and be in the reactor coolant pressure boundary or the high pressure portion of the system. -----</p>	<p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<p><u>AND</u></p> A.2 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each SI cold leg injection line and each RHR RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.</p>	<p>24 months</p> <p><u>AND</u></p> <p>Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each SI hot leg injection line RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.</p>	<p>40 months</p> <p><u>AND</u></p> <p>Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. One containment sump A monitor (level or pump actuation); and
- b. One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1.1 Perform SR 3.4.13.1. <u>OR</u>	Once per 24 hours
	A.1.2 Verify containment air cooler condensate collection system is OPERABLE.	24 hours
	<u>AND</u> A.2 Restore required containment sump monitor to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required containment atmosphere radioactivity monitor inoperable.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>B.1.1 Analyze grab samples of the containment atmosphere.</p> <p style="text-align: center;"><u>OR</u></p> <p>B.1.2 Perform SR 3.4.13.1.</p> <p style="text-align: center;"><u>AND</u></p> <p>B.2 Restore required containment atmosphere radioactivity monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>Once per 24 hours</p> <p>30 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required containment sump monitor inoperable.</p> <p><u>AND</u></p> <p>Particulate containment atmosphere radioactivity monitor inoperable.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>C.1.1 Analyze grab samples of the containment atmosphere.</p> <p><u>OR</u></p> <p>C.1.2 Perform SR 3.4.13.1</p> <p><u>AND</u></p> <p>C.2.1 Restore required containment sump monitor to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2.2 Restore particulate containment atmosphere radioactivity monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>Once per 24 hours</p> <p>30 days</p> <p>30 days</p>
<p>D. Required Action and associated Completion Time of Conditions A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
<p>E. All required monitors inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitor.	92 days
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment sump monitor.	24 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity not within limit.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. <u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 8 hours 7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.	B.1 Be in MODE 3 with $T_{avg} < 500^\circ\text{F}$.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Gross specific activity not within limit.	C.1 Be in MODE 3 with $T_{avg} < 500^{\circ}F$.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.16.1 Verify reactor coolant gross specific activity $\leq 100/E \mu Ci/gm$.	7 days
SR 3.4.16.2 -----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 1.0 \mu Ci/gm$.	14 days <u>AND</u> Between 2 and 10 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.16.3 -----NOTE----- Only required to be performed in MODE 1. ----- Determine \bar{E} from a reactor coolant sample.</p>	<p>Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for \geq 48 hours.</p> <p><u>AND</u></p> <p>Every 184 days thereafter</p>

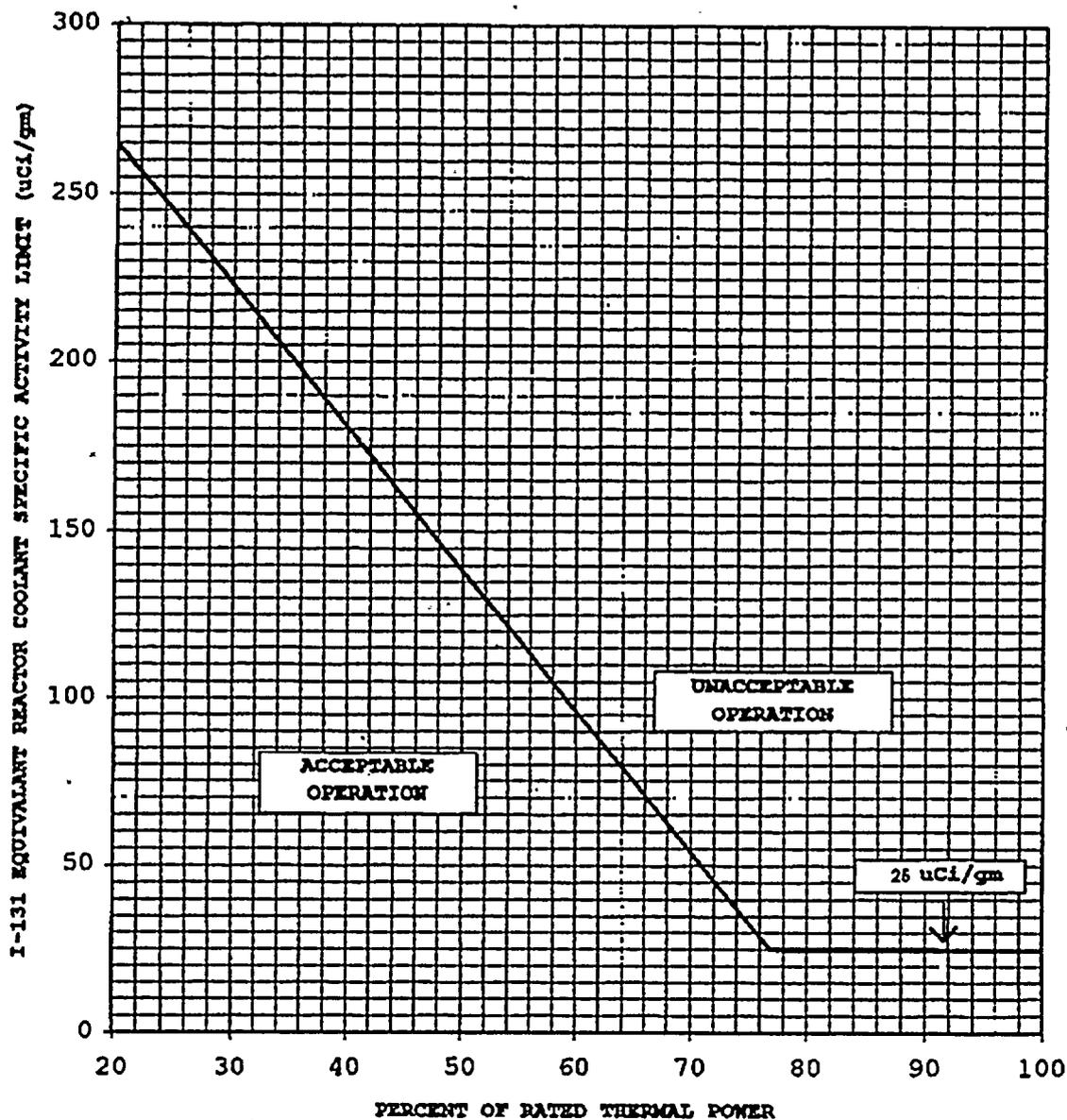


Figure 3.4.16-1
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) design criterion will be met for each of the transients analyzed.

The design method employed to meet the DNB design criterion for fuel assemblies is the Improved Thermal Design Procedure (ITDP). With the ITDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, ITDP design limit departure from nucleate boiling ratio (DNBR) values are determined in order to meet the DNB design criterion.

The ITDP design limit DNBR values are 1.34 and 1.33 for the typical and thimble cells, respectively, for fuel analyses with the WRB-2 correlation.

Additional DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR values are 1.52 and 1.51 for the typical and thimble cells, respectively.

(continued)

BASES

BACKGROUND
(continued)

For both the WRB-1 and WRB-2 correlations, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

The RCS pressure limit as specified in the COLR, is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit as specified in the COLR, is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate as specified in the COLR, normally remains constant during an operational fuel cycle with both pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNB design criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the plant that could impact these parameters must be assessed for their impact on the DNB design criterion. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The limit for pressurizer pressure is based on a ± 30 psig instrument uncertainty. The accident analyses assume that nominal pressure is maintained at 2235 psig. By Reference 2, minor fluctuations are acceptable provided that the time averaged pressure is 2235 psig.

The RCS coolant average temperature limit is based on a $\pm 4^\circ\text{F}$ instrument uncertainty which includes a $\pm 1.5^\circ\text{F}$ deadband. It is assumed that nominal T_{avg} is maintained within $\pm 1.5^\circ\text{F}$ of 573.5°F . By Reference 2, minor fluctuations are acceptable provided that the time averaged temperature is within 1.5°F of nominal.

The limit for RCS flow rate is based on the nominal T_{avg} and SG plugging criteria limit. Additional margin of approximately 3% is then added for conservatism.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

(continued)



BASES

LCO
(continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNB design criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In MODE 2, an increased DNBR margin exists. In all other MODES, the power level is low enough that DNB is not a concern.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

(continued)

BASES

ACTIONS

A.1 (continued)

The 2 hour Completion Time for restoration of the parameters provides sufficient time to determine the cause for the off normal condition, to adjust plant parameters, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

Measurement of RCS total flow rate once every 24 months verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. This verification may be performed via a precision calorimetric heat balance or other accepted means.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. Verification of RCS flow rate on a shorter interval is not required since this parameter is not expected to vary during steady state operation as there are no RCS loop isolation valves or other installed devices which could significantly alter flow. Reduced performance of a reactor coolant pump (RCP) would be observable due to bus voltage and frequency changes, and installed alarms that would result in operator investigation.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the plant in the best condition for performing the SR. The Note states that the SR shall be performed within 7 days after reaching 95% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 95% RTP to obtain the stated RCS flow accuracies.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 15.
 2. NRC Memorandum from E.L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operations Inspection to Distribution; Subject: "Discussion of Licensed Power Level (AITS F14580H2)," dated August 22, 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the RCS water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures greater than or equal to the HZP temperature of 547°F. The minimum temperature for criticality limitation provides a small band, 7°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1, and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

(continued)

BASES

APPLICABILITY
(continued)

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at $\leq 5\%$ RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO. The need to perform the PHYSICS TESTS to ensure that the operating characteristics of the core are consistent with design predictions provides sufficient justification to allow a temporary decrease in the RCS minimum temperature for criticality limit.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $K_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period due to the proximity to MODE 2 conditions. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $K_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.1

This SR verifies that RCS T_{avg} in each loop is $\geq 540^{\circ}F$ within 30 minutes prior to achieving criticality. This ensures that the minimum temperature for criticality is being maintained just before criticality is reached.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.2 (continued)

RCS loop average temperature is required to be verified at or above 540°F every 30 minutes in MODE 1, and in MODE 2 with $k_{eff} \geq 1.0$. The 30 minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated.

This SR is modified by a Note that only requires the SR to be performed if any RCS loop T_{avg} is $< 547^\circ\text{F}$ and the low T_{avg} alarm is either inoperable or not reset. The T_{avg} alarm provides operator indication of low RCS temperature without requiring independent verification while a $T_{avg} > 547^\circ\text{F}$ in both RCS loops is within the accident analysis assumptions. If the T_{avg} alarm is to be used for this SR, it should be calibrated consistent with industry standards.

This surveillance is replaced by SR 3.1.8.2 during PHYSICS TESTING.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

(continued)

BASES

BACKGROUND
(continued)

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material has been established by periodically removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves have been adjusted based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be $\geq 40^\circ\text{F}$ above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

(continued)

BASES

BACKGROUND
(continued)

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE
SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and result in nonductile failure of the RCPB which is an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

- a. The limit curves for heatup, cooldown, and ISLH testing; and
- b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

(continued)

BASES

LCO
(continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and
- c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

(continued)

BASES

APPLICABILITY
(continued)

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

(continued)

BASES

ACTIONS

A.1 (continued)

Condition A is modified by a Note stating that Required Action A.2 shall be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event which is best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished quickly in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1 (continued)

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994.
 2. 10 CFR 50, Appendix G.
 3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
 4. ASTM E 185-82, July 1982.
 5. 10 CFR 50, Appendix H.
 6. Regulatory Guide 1.99, Revision 2, May 1988.
 7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODE 1 > 8.5% RTP

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;
- b. Improving the neutron economy by acting as a reflector;
- c. Carrying the soluble neutron poison, boric acid; and
- d. Providing a second barrier against fission product release to the environment.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds all possible instrumentation errors (Ref. 2). The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with both RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. Adequate heat transfer between the reactor coolant and the secondary side is ensured by maintaining $\geq 16\%$ SG level in accordance with LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," which provides sufficient water inventory to cover the SG tubes.

RCS Loops - MODE 1 > 8.5% RTP satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required to be in operation at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY In MODE 1 > 8.5% RTP, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, both RCS loops are required to be OPERABLE and in operation in this MODE to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower MODES as indicated by the LCOs for MODES 1 \leq 8.5% RTP, 2, 3, 4, and 5.

Operation in other MODES is covered by:
LCO 3.4.5, "RCS Loops - MODES 1 \leq 8.5% RTP, 2, AND 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft" (MODE 6);
and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 1 < 8.5% RTP. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

(continued)

BASES

ACTIONS

A.1 (continued)

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

REFERENCES

1. UFSAR, Chapter 15.
 2. UFSAR, Section 15.0.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODES 1 \leq 8.5% RTP, 2, AND 3

BASES

BACKGROUND

In MODE 1 \leq 8.5% RTP, and in MODE 2 and 3, the primary function of the RCS is the removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant. The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission (MODE 1 \leq 8.5% RTP and MODE 2 only);
- b. Improving the neutron economy by acting as a reflector (MODE 1 \leq 8.5% RTP and MODE 2 only);
- c. Carrying the soluble neutron poison, boric acid; and
- d. Providing a second barrier against fission product release to the environment.

The reactor coolant is circulated through two RCS loops, connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 1 \leq 8.5% RTP and MODE 2, the RCPs are used to provide forced circulation of the reactor coolant to ensure mixing of the coolant for proper boration and chemistry control and to remove the limited amount of reactor heat. In MODE 3, the RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 1 \leq 8.5% RTP, 2, and 3 reactor and decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). In MODE 1 \leq 8.5% RTP, and in MODES 2 and 3, these analyses include evaluation of main steam line breaks and uncontrolled rod withdrawal from a subcritical condition. The most limiting accident with respect to DNB limits for MODES 1 \leq 8.5% RTP, 2, and 3 is a main steam line break. This is due to the potential for recriticality and because of the high hot channel factors that may exist if the most reactive control rod is stuck in its fully withdrawn position.

A main steam line break has been analyzed for both the case with one and two RCS loops in operation at hot zero power (HZP) conditions with acceptable results (Ref. 1). However, with only one RCS loop in operation and offsite power available, additional shutdown margin is required since the reduced flow produces an adverse effect on DNB limits.

The startup of an inactive reactor coolant pump (RCP) up to 8.5% RTP has been evaluated and found to result in only limited power and temperature excursions that are bounded by a main steam line break with only one RCS Loop in operation (Refs. 2 and 3).

Analyses have also been performed which demonstrate that reactor heat greater than 5% RTP can be removed by natural circulation alone (Ref. 4).

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. RCS Loops - MODES 1 \leq 8.5 % RTP, 2, and 3 satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that both RCS loops be OPERABLE. Only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS up to 8.5% RTP. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met. Requiring one RCS loop in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

The Note permits all RCPs to be de-energized for \leq 1 hour per 8 hour period in MODE 3. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test was satisfactorily performed during the initial startup testing program (Ref. 5). If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

The no flow test may be performed in MODE 3, 4, or 5. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

(continued)

BASES

LCO
(continued)

- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and able to provide forced flow if required.

APPLICABILITY

In MODES 1 \leq 8.5% RTP, 2, and 3, this LCO ensures forced circulation of the reactor coolant to remove reactor and decay heat from the core and to provide proper boron mixing.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
 - LCO 3.4.6, "RCS Loops - MODE 4";
 - LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
 - LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
 - LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft" (MODE 6);
and
 - LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).
-

ACTIONS

A.1 and A.2

If one RCS loop is inoperable, redundancy for heat removal is lost. The Required Actions are to verify that the SDM is within limits specified in the COLR. This action is required to ensure that adequate SDM exists in the event of a main steam line break with only one RCS loop in operation. The 12 hour Frequency considers the time required to obtain RCS boron concentration samples and the low probability of a main steam line break during this time period.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The inoperable RCS loop must be restored to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the reactor and decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one RCS loop is inoperable. This allowance is provided because a single RCS loop can provide the required cooling to remove reactor and decay heat consistent with safety analysis assumptions.

B.1

If restoration of the inoperable loop is not possible within 72 hours, the plant must be brought to MODE 4. In MODE 4, the plant may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1, C.2, and C.3

If two RCS loops are inoperable, or no RCS loop is in operation, except during conditions permitted by the Note in the LCO section, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that each required RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of the control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is \geq 16% for two RCS loops. If the SG secondary side narrow range water level is $<$ 16%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of reactor or decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.5.3

Verification that the required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR Section 15.1.5.
 2. UFSAR Section 15.4.3.
 3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-9, Startup of an Inactive Loop, R. E. Ginna," dated August 26, 1981.
 4. UFSAR Sections 14.6.1.5.6 and 15.2.5.2.
 5. UFSAR Section 14.6.1.5.5.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the cladded fuel. The SGs or the RHR heat exchangers provide the heat sink. The RCPs and the RHR pumps circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCS or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCS or one RHR loop for decay heat removal and transport. The flow provided by one RCS loop or one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 have been identified in the NRC Policy Statement as important contributors to risk reduction.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs and RHR pumps to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program was the validation of rod drop times during cold conditions, both with and without flow (Ref. 1). If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

(continued)

BASES

LCO
(continued)

Note 2 requires that the pressurizer water volume be < 324 cubic feet (38% level), or that the secondary side water temperature of each SG be $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed condition.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2. RCPs are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. An OPERABLE RHR loop may be isolated from the RCS provided that the loop can be placed into service from the control room. RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

(continued)

BASES

APPLICABILITY
(continued)

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
 - LCO 3.4.5, "RCS Loops - MODES 1 ≤ 8.5% RTP, 2, AND 3";
 - LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
 - LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
 - LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
 - LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).
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ACTIONS

A.1

If one RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. If no RHR is available, the plant cannot enter a reduced MODE since no long term means of decay heat removal would be available. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one RHR loop is inoperable and both RCS loops are inoperable, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS

B.1 (continued)

Required Action B.1 is modified by a Note stating that only the Required Actions of Condition C are entered if all RCS and RHR loops are inoperable. With all RCS and RHR loops inoperable, MODE 5 cannot be entered and Required Actions C.1 and C.2 are the appropriate remedial actions.

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.6.2

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq 16\%$. If the SG secondary side narrow range water level is $< 16\%$, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 14.6.1.2.6.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat either to the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is normally circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining one SG with a secondary side water level at or above 16% to provide an alternate method for decay heat removal.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or one SG with a secondary side water level $\geq 16\%$. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is one SG with a secondary side water level $\geq 16\%$. Should the operating RHR loop fail, the SG could be used to remove the decay heat.

Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program was the validation of rod drop times during cold conditions, both with and without flow (Ref. 1). If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

(continued)

BASES

LCO
(continued)

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period ≤ 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the pressurizer water volume be < 324 cubic feet (38% level), or that the secondary side water temperature of each SG be $\leq 50^\circ\text{F}$ above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature are to prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed Condition.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops. A planned heatup is a scheduled transition to MODE 4 within a defined time period.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it is OPERABLE in accordance with the Steam Generator Tube Surveillance Program, with the minimum water level specified in SR 3.4.7.2.

(continued)

BASES (continued)

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The RCS loops are considered filled until the isolation valves are opened to facilitate draining of the RCS. The loops are also considered filled following the completion of filling and venting the RCS. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be $\geq 16\%$.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 $> 8.5\%$ RTP";
- LCO 3.4.5, "RCS Loops - MODES 1 $\leq 8.5\%$ RTP, 2, AND 3";
- LCO 3.4.6, "RCS Loops - MODE 4";
- LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
- LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and both SGs have secondary side water levels $< 16\%$, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore at least one SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until an RHR loop is restored to OPERABLE status or SG secondary side water level is restored.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

This SR requires verification of SG OPERABILITY. Verifying that at least one SG is OPERABLE by ensuring its secondary side narrow range water level is $\geq 16\%$ ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby RHR pump. If secondary side water level is $\geq 16\%$ in at least one SG, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 14.6.1.2.6
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE
SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation to transfer heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one operating RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

(continued)

BASES

LCO
(continued)

Note 1 permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and requires that the following conditions be met:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation;
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and
- c. No draining operations are permitted that would further reduce the RCS water volume and possibly cause a more rapid heatup of the remaining RCS inventory.

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. The RCS loops are considered not filled from the time period beginning with the opening of isolation valves and draining of the RCS and ending with the completion of filling and venting the RCS.

(continued)

BASES

APPLICABILITY
(continued)

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
 - LCO 3.4.5, "RCS Loops - MODES 1 \leq 8.5% RTP, 2, AND 3";
 - LCO 3.4.6, "RCS Loops - MODE 4";
 - LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
 - LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft" (MODE 6);
and
 - LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).
-

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until the second RHR loop is restored to OPERABLE status.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level and the required heater capacity. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of this LCO is to ensure that a steam bubble exists in the pressurizer prior to, and during, power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensable gases are typically present in the RCS and can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control. These noncondensable gases can be ignored if the steam bubble is present.

(continued)

BASES

BACKGROUND
(continued)

This LCO also ensures that adequate heater capacity is available in the pressurizer to support natural circulation following an extended loss of offsite power. Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. These heaters are divided into two groups, a control/variable group and a backup group. The control/variable group is normally used during power operation since these heaters have inverse proportional control with respect to the pressurizer pressure. The backup group is either fully on or off with setpoints that are below those for the control/variable group. Both groups of heaters receive power from the Engineered Safety Feature (ESF) 480 V buses, however, the heaters are shed following a loss of offsite power or safety injection signal. The heaters can be manually loaded onto the diesel generators if required.

A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained during natural circulation. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat. Unless adequate heater capacity is available, the required subcooling margin in the primary system cannot be maintained. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat. Maintaining necessary subcooled margin during normal power operation is controlled by meeting the requirements for pressurizer level and LCO 3.4.1, "RCS Pressure, Temperature and Flow Departure From Nucleate Boiling (DNB) Limits."

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting with respect to pressurizer parameters. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

The maximum pressurizer water level limit ensures that a steam bubble exists and satisfies Criterion 2 of the NRC Policy Statement.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation, however, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO. The pressurizer heaters are assumed to be available within one hour following the loss of offsite power and initiation of natural circulation (Ref. 3).

LCO

The LCO establishes the minimum conditions required to ensure that a steam bubble exists within the pressurizer and that sufficient heater capacity is available to support an extended loss of offsite power event. For the pressurizer to be considered OPERABLE, the limits established in the SRs for water level and heater capacity must be met and the heaters must be capable of being powered from an emergency power source within one hour.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3 to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

(continued)



BASES

APPLICABILITY
(continued)

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply (Ref. 4). In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

If the pressurizer water level is > 650 cubic feet, which is equivalent to 87%, the ability to maintain a steam bubble may no longer exist. The steam bubble is necessary to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit is the same as the Pressurizer High Level Trip.

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the plant must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the pressurizer heaters capacity is < 100 KW, the ability to maintain RCS pressure to support natural circulation may no longer exist. By maintaining RCS pressure control, a margin to subcooling is provided. The value of 100 KW is based on the amount needed to support natural circulation after accounting for heat losses through the pressurizer insulation during an extended loss of offsite power event.

If the capacity of the pressurizer heaters is not within the limit, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

This SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power required. This may be done by testing the power supply output by verifying the electrical load on Buses 14 and 16 with the respective heater groups on and off. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 15.
 2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
 3. Letter from B. L. King, Westinghouse Electric Corporation, to R. C. Mecredy, RG&E, Subject: "Ability to Maintain Subcooled Conditions During an Extended Loss of Offsite Power," dated September 26, 1979.
 4. Letter from D. M. Crutchfield, NRC; to L. D. White, Jr. RG&E, Subject: "Lessons Learned Category 'A' Evaluation," dated July 7, 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 288,000 lbm/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5 and in MODE 6 with reactor vessel head on; however, in MODE 4, with either RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR, and MODE 5 and MODE 6 with the reactor vessel head on and the SG primary system manway and pressurizer manway closed and secured in position, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the $\pm 1\%$ tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

(continued)

BASES

BACKGROUND
(continued)

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the locked rotor accident which remains below 120% of the design pressure consistent with the original maximum transient pressure limit for the RCS (Refs. 2, 3 and 4). The consequences of exceeding the American Society of Mechanical Engineers (ASME) and USAS Section B31.1 pressure limits (Refs. 1 and 4) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE
SAFETY ANALYSES

All accident and safety analyses in the UFSAR (Ref. 5) that require safety valve actuation assume operation of both pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 6) is also based on operation of both safety valves. Accidents that could result in overpressurization if not properly terminated include:

- a. Uncontrolled rod withdrawal from full power;
- b. Loss of reactor coolant flow;
- c. Loss of external electrical load (including the complete loss of steam flow to the turbine);
- d. Loss of normal feedwater;
- e. Loss of all AC power to station auxiliaries; and
- f. Locked rotor.

Detailed analyses of the above transients are contained in Reference 5. Safety valve actuation is required in events c, d, e, and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits following testing are based on the $\pm 1\%$ tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The OPERABILITY limits of + 2.4%, - 3% are based on the analyzed events. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure for all transients except locked rotor accidents which has an allowed limit of 120% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when either RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned or the SG primary system manway or the pressurizer manway open.

(continued)

BASES (continued)

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with either RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperature at or below the LTOP enable temperature specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer surges, and thereby removes the need for overpressure protection by both pressurizer safety valves.

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 7), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is + 2.4%, - 3% for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1 (continued)

This SR is modified by a Note that allows entry into MODES 3 and 4 without having performed the SR for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition until completion of the surveillance.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
 2. UFSAR, Section 15.3.2.
 3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-1, XV-2, XV-3, XV-4, XV-5, XV-6, XV-7, XV-8, XV-10, XV-12, XV-14, XV-15, and XV-17, Design Basis Events, Accidents, and Transients (R.E. Ginna)," dated September 4, 1981.
 4. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967 edition.
 5. UFSAR, Chapter 15.
 6. WCAP-7769, "Topical Report, Overpressure Protection for Westinghouse Pressurized Water Reactors," Rev. 1, June 1972.
 7. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs (430 and 431C) are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Motor operated block valves (515 and 516), which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater and auxiliary feedwater. The PORVs are also used to mitigate the effects of an anticipated transient without scram (ATWS) event which is also not within the design basis.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The two PORVs (in manual operation only) and their associated block valves are powered from two separate safety trains.

(continued)

BASES

BACKGROUND
(continued)

The plant has two PORVs, each having a relief capacity of 179,000 lb/hr at 2335 psig. The PORVs are normally opened by using instrument air which is supplied through separate solenoid operated valves (8620A and 8620B). The safety related source of motive air is from two separate nitrogen accumulators that are normally isolated from the PORVs by solenoid operated valves 8619A and 8619B; however, solenoid operated valves 8620A and 8620B must be in the vent position to close the PORVs regardless of which motive air source is used.

The functional design of the PORVs is based on maintaining pressure below the pressurizer high pressure reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE
SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also used in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the pressurizer high pressure trip and pressurizer safety valve setpoints; thus the DNBR calculation is more conservative assuming the same initial RCS temperature since the pressurizer pressure is limited. Events that assume this condition include a loss of external electrical load and other transients which result in a decrease in heat removal by the secondary system (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation by the nitrogen accumulators to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV is required to be OPERABLE to mitigate the effects associated with an SGTR and its block valve must be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to automatically open with a subsequent failure to close. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high.

The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves. Therefore, the LCO is applicable in MODES 1, 2, and 3.

The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

(continued)

BASES (continued)

ACTIONS

Note 1 has been added to clarify that both pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis) for Condition A. Note 2 has been added to clarify that both block valves are treated as separate entities, each with separate Completion Times, for Condition C. The exception for LCO 3.0.4, Note 3, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES due to LTOP considerations.

A.1 and A.2

With the PORVs OPERABLE and not capable of being automatically controlled, either the PORVs must be restored or the flow path isolated within 1 hour. Although a PORV may not be capable of being automatically controlled, it may be able to be manually opened and closed, and therefore, able to perform its function. A PORV is considered not capable of being automatically controlled for any problem which prevents the PORV from automatically closing once it has automatically opened. This may be due to instrumentation problems. Not capable of automatic control does not include problems which only prevent the PORV from automatically opening (e.g., loss of instrument air to the PORV). It also does not include problems which prevent the PORV from both automatically opening and automatically closing. For these reasons, the block valve may either be closed to isolate the flowpaths or isolated by placing the PORV control switch in the closed position. However, if the block valve is closed to isolate the flowpath, the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2). Seat leakage problems are controlled by LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is not capable of being manually cycled, it is inoperable and must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. PORV inoperability includes (but is not limited to) the inability of the solenoid operated isolation valve from the nitrogen accumulator to open or the solenoid operated isolation valve from instrument air to vent. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is a second PORV that is OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition E.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. Manual control is accomplished by placing the PORV control board switch in the closed position. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because the PORV is not capable of automatically opening and the small potential for an SGTR or other event requiring Manual operation, the operator is permitted a Completion Time of 7 days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is limited to 7 days since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of 7 days, the PORV will again be capable of automatically responding to an overpressure event, and the block valves capable of isolating a stuck open PORV which may result from the overpressure event. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition E.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If both block valves are inoperable, then it is necessary to either restore at least one block valve to OPERABLE status within the Completion Time of 1 hour or place the PORVs in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valves cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORVs in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. Manual control is accomplished by placing the PORV control board switch in the closed position. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because the PORV is not capable of automatically opening and the small potential for an SGTR or other event requiring Manual operation, the operator is permitted a Completion Time of 72 hours to restore at least one inoperable block valve to OPERABLE status. The time allowed to restore one block valve is limited to 72 hours since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If at least one block valve is restored within the Completion Time of 72 hours, at least one PORV will again be capable of automatically responding to an overpressure event, and the associated block valve capable of isolating a stuck open PORV which may result from the overpressure event. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition E.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

If the Required Action of Condition A, B, C, or D is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1, F.2, F.3, and F.4

If both PORVs are not capable of being manually cycled, they are inoperable and it is necessary to initiate action to restore one PORV to OPERABLE status immediately since no relief valve is available to mitigate the effects associated with an SGTR. Therefore, operators must either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

(continued)

BASES

ACTIONS

F.1, F.2, F.3, and F.4 (continued)

If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE which does not require manual PORV operation. To achieve this status, the plant must be brought to MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 8 hours. In MODE 3 with the RCS average temperature $< 500^{\circ}\text{F}$, the saturation pressure of the reactor coolant is below the setpoint of the main steam safety valves. Since the RWST contains a larger volume of water than the secondary side of an SG, the leak through the ruptured tube will stop after the SG is filled to capacity. Therefore, an SGTR can be mitigated under these conditions without any release of radioactive fluid through the main steam safety valves. Entering a lower MODE is not desirable with both PORVs inoperable and not capable of being manually cycled since the PORVs are also required for low temperature overpressure protection. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 2). If the block valve is closed to isolate a PORV that is OPERABLE and is not leaking in excess of the limits of LCO 3.4.13, "RCS Operational LEAKAGE," then opening the block valve is necessary to verify that the PORV can be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed per LCO 3.4.13. This prevents the need to open the block valve when the associated PORV is leaking > 10 gpm creating the potential for a plant transient.

SR 3.4.11.2

This SR requires a complete cycle of each PORV using the nitrogen accumulators. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. UFSAR, Section 15.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The LTOP system also protects the RHR system from overpressurization during the RHR mode of operation. The PTLR provides the maximum allowable actuation logic setpoints for the pressurizer power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

(continued)

BASES

BACKGROUND
(continued)

This LCO provides RCS overpressure protection by restricting coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the Emergency Core Cooling System (ECCS) accumulators and rendering all safety injection (SI) pumps incapable of RCS injection when the PORVs provide the RCS vent path and rendering a minimum of two SI pumps incapable of RCS injection when the RCS is depressurized with an RCS vent ≥ 1.1 square inches. The pressure relief capacity requires either two redundant PORVs or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

By restricting coolant input capability, the ability to provide core coolant addition is minimized. The LCO does not require the makeup control system to be deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If the conditions require the use of SI for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The two redundant PORVs or a depressurized RCS with an open RCS vent is also sufficient to protect the RHR system during the RHR mode of operation for events which cause an increase in system pressure.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure exceeds the limit selected to prevent a condition that is not within the acceptable region provided in the PTLR. The PORVs are opened by coincident actuation of two-of-three RCS pressure channels. The PTLR presents the PORV setpoint for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and then reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

(continued)

BASES

BACKGROUND
(continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV's internals or blocking it open, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent path. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE
SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits for all Design Basis Accidents. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding the LTOP enable temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At or below the LTOP enable temperature specified in the PTLR, overpressure prevention requires two OPERABLE PORVs or a depressurized RCS and a sufficiently sized RCS vent. Each of these overpressure protection systems has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases as a result of neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 3 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

- a. Inadvertent safety injection (SI); or
- b. Charging/letdown flow mismatch.

Heat Input Type Transients

- a. Inadvertent actuation of pressurizer heaters;
- b. Loss of RHR cooling; or
- c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

Analyses have determined that the mass input transients are the bounding case for overpressurization of the RCS (Ref. 3). The two categories of mass input transients were analyzed with respect to utilizing a single PORV or an RCS vent ≥ 1.1 square inches as overpressure protection. The inadvertent actuation of a single SI pump provides a larger mass addition to the RCS than isolation of letdown with all three charging pumps operating. A single PORV was determined to be incapable of mitigating the overpressure transient resulting from actuation of a SI pump, but is capable of mitigating the charging/letdown mismatch transient. An RCS vent ≥ 1.1 square inches can mitigate both the inadvertent SI and charging/letdown flow mismatch transients.

Therefore, the following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. Rendering all SI pumps incapable of injection into the RCS when the PORVs provide the RCS vent path and rendering all but one SI pump incapable of injection into the RCS when the RCS is depressurized with an RCS vent of ≥ 1.1 square inches;
- b. Deactivating the ECCS accumulator discharge motor operated isolation valves in their closed positions; and
- c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop or pressurizer level $\geq 38\%$. LCO 3.4.6, "RCS Loops-MODE 4," and LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," provide this protection.

The Reference 3 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits with the maximum allowed coolant input capability. Since neither one PORV nor the RCS vent can handle the pressure transient produced from ECCS accumulator injection when RCS temperature is low, the LCO also requires the ECCS accumulators isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated ECCS accumulators must have their discharge valves closed and the valve power supply removed. The analyses show the effect of ECCS accumulator discharge is over a narrower RCS temperature range (200°F and below) than that of the LCO. Fracture mechanics analyses established the temperature of LTOP Applicability at the LTOP enable temperature specified in the PTLR.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having procedures to manually establish makeup capability.

The events which potentially overpressurize the RHR system during the RHR mode of operation are included within the mass and heat input transients analyzed for LTOP conditions. Therefore, an OPERABLE LTOP System ensures that the RHR system will not be overpressurized during the RHR mode of operation.

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BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient for the PORVs of a charging/letdown flow mismatch. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met and that the RHR system will not be overpressurized.

The PORV setpoints in the PTLR are updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 1.1 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, which maintains RCS pressure less than the maximum pressure on the P/T limit curve. The limiting transient for this LTOP configuration is an SI actuation with one SI pump OPERABLE.

An RCS vent \geq 1.1 square inches with the RCS depressurized also prevents overpressurization of the RHR system.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of the NRC Policy Statement.

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires the ECCS accumulators to be isolated. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," defines SI actuation OPERABILITY for the LTOP MODE 4 small break LOCA.

The elements of the LCO that provide low temperature overpressure mitigation are:

- a. Two OPERABLE PORVs and no SI pump capable of injecting into the RCS.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the valve and its control circuits.

- b. A depressurized RCS and an RCS vent and a maximum of one SI pump capable of injecting into the RCS.

An RCS vent is OPERABLE when open with an area of ≥ 1.1 square inches.

(continued)

BASES

LCO
(continued)

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The LCO is modified by two Notes. The first Note allows performance of the secondary side hydrostatic tests without the PORVs and RCS vent OPERABLE; however no SI pump may be capable of injecting into the RCS during this test. This exclusion is necessary since a pressure differential of ≤ 800 psid is maintained between the primary and secondary sides during the test. This restricted pressure differential limits the stresses placed on the SG which can cause cladding in the primary channel to separate from the base metal and result in the need for difficult repairs in a high radiation area. To maintain this pressure differential limit, RCS pressure must be increased above the PORV setpoint for LTOP conditions. The test cannot be performed above the LTOP enable temperature since the steam lines may not be able to accommodate the associated thermal expansion if they are heated. Therefore, all three SI pumps must be incapable of injecting into the RCS during these secondary side hydrostatic tests (Ref. 6).

The second Note only requires an ECCS accumulator to be isolated when the accumulator pressure is greater than or equal to the maximum pressure for the existing RCS cold leg temperature allowed in the PTLR. Accumulator pressure below this limit will not overpressurize the RCS beyond analyzed conditions. The accumulator is isolated when the discharge motor operated valve is closed and its associated power supply is removed.

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or the RHR system is in the RHR operating mode, in MODE 5 when the SG primary system manway and pressurizer manway are closed and secured in position, and in MODE 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enable temperature specified in the PTLR. When the reactor vessel head is off or the SG primary system manway or pressurizer manway are open, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above the LTOP enable temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A.1

With one or more SI pumps capable of injecting into the RCS and the PORVs provide the RCS vent path, RCS overpressurization is possible. To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of taking action to remove the RCS from this potential condition.

Condition A is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

(continued)

BASES

ACTIONS

B.1 (continued)

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

Condition B is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

C.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PORVs inoperable in MODE 5 with the SG primary system manway and pressurizer manway closed and secured in position, or in MODE 6 with the head on and the SG primary system manway and pressurizer manway closed and secured in position, the PORV must be restored to OPERABLE status in 72 hours. Restoring the PORV to OPERABLE status provides required redundancy.

The Completion Time of 72 hours to restore the PORV to OPERABLE status represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one PORV to protect against overpressure events.

Condition C is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

(continued)

BASES

ACTIONS
(continued)

D.1

With two or more SI pumps capable of injecting into the RCS and the RCS is depressurized with an RCS vent of ≥ 1.1 square inches, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of taking action to the RCS from this potential condition.

Condition D is modified by a Note which states that this condition is only applicable to LCO 3.4.12.b (i.e., when there is a RCS vent path ≥ 1.1 square inches.

E.1, F.1, and F.2

An unisolated ECCS accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action F.1 and Required Action F.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to greater than the LTOP enable temperature specified in the PTLR, a maximum accumulator pressure of 800 psig (relief valve setpoint) cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

G.1 and G.2

At least one charging pump must be in the pull-stop position within 1 hour and the RCS must be depressurized and a vent must be established within 8 hours when:

- a. Both required PORVs are inoperable; or

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

- b. A Required Action and associated Completion Time of Condition A, B, C, or F is not met; or
- c. The LTOP System is inoperable for any reason other than Condition A, B, C, or E.

The Completion Time of one hour to restrict the coolant input capability to the RCS considers the relatively low probability of an overpressure event during this time period and provides the operator time to render a charging pump incapable of injecting by placing it in the pull-stop position. Only one disabling device is required since there is a relatively small probability of an inadvertent charging pump actuation during the 8 hours before RCS depressurization is achieved and a vent established. The disabling of a charging pump is necessary since RV 203 cannot mitigate a charging/letdown mismatch event if RHR is providing decay heat removal above MODE 5 and three charging pumps are operating.

The vent must be sized ≥ 1.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel and to protect the RHR system from overpressurization.

The Completion Time of 8 hours to depressurize the RCS and establish a vent considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SI pumps must be verified incapable of injecting into the RCS when the PORVs provide the RCS vent path (LCO 3.4.12.a) and a minimum of two SI pumps must be verified incapable of injecting into the RCS when the RCS is depressurized and an RCS vent ≥ 1.1 square inches is established (LCO 3.4.12.b). The SI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the following:

- a. placing the pump control switch in the pull-stop position and closing at least one valve in the discharge flow path;
- b. locking closed a manual isolation valve in the injection path; or
- c. closing a motor operated isolation valve in the injection path and removing the AC power source.

The flowpaths through the test connections associated with the ECCS accumulator check valves (i.e., lines containing air operated valves 839A, 839B, 840A, and 840B) and the ECCS accumulator fill lines (i.e., lines containing air operated valves 835A and 835B) do not have to be isolated for this SR since the potential mass addition from a single SI pump through these six lines is limited by the installed orifices to less than that assumed for the charging/letdown mismatch analysis.

(continued)

D
BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

The ECCS accumulator motor operated isolation valves can be verified closed by use of control board indication for valve position. This verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. If the accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the PORV setpoint.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The Frequency of every 12 hours thereafter for SR 3.4.12.3 ensures that the ECCS accumulator motor operated isolation valves are maintained closed and do not result in a potential LTOP actuation.

SR 3.4.12.4

The RCS vent of ≥ 1.1 square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that cannot be locked.
- b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.b.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve may be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a CHANNEL OPERATIONAL TEST (COT) is required every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is therefore not required.

A Note has been added indicating that this SR is required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR if it has not been performed within the previous 31 days. Depending on the cooldown rate, the COT may not have been performed before entry into the LTOP MODES. The test must be performed within 12 hours after entering the LTOP MODES. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.12.7

Verification once within 12 hours and every 31 days thereafter that power is removed from each ECCS accumulator motor operated isolation valve ensures that at least two independent actions must occur before the accumulator is capable of injecting into the RCS. Since power is removed under administrative control and valve position is verified every 12 hours, the performance of this surveillance once within 12 hours and every 31 days thereafter will provide assurance that power is removed.

This SR is modified by a Note which states that the Surveillance is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR. If the accumulator pressure is below this limit, the LTOP limit cannot be exceeded and the surveillance is not required.

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11, "NRC Position on Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations."
3. UFSAR, Section 5.2.2.
4. 10 CFR 50, Section 50.46.
5. 10 CFR 50, Appendix K.

(continued)

BASES

REFERENCES
(continued)

6. Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18," dated July 26, 1979.
 7. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

Atomic Industry Forum (AIF) GDC 16 (Ref. 1) requires that means be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (RCPB). AIF-GDC 34 also requires that the RCPB be designed to reduce the probability of rapid propagation failures. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. The leakage detection systems support these requirements by both detecting RCS LEAKAGE and identifying the location of its source. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the plant and the public.

(continued)

BASES

BACKGROUND
(continued)

A limited amount of leakage inside containment is expected from auxiliary systems (e.g. component cooling water) that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE
SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event (Ref. 2).

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 0.5 gpm primary to secondary LEAKAGE as the initial condition. The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The assumed 0.5 gpm primary to secondary LEAKAGE is relatively inconsequential.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The SLB outside of containment is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 0.5 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident outside of containment are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). However, a lower LEAKAGE limit is assumed for all SLBs to prevent a coincident SGTR due to the large stresses placed on the SG tubes as a result of the rapid cooldown and depressurization. These stress calculations conservatively assume a tube with a 0.4 inch long through-wall crack in a location with 40% local wall thinning. The analyses demonstrate that the integrity of the selected tube is maintained with sufficient margin after the SLB. The assumed through-wall crack of 0.4 inches corresponds to 0.1 gpm leakage under normal operating conditions (Ref. 4). Therefore, the primary to secondary LEAKAGE is limited to 0.1 gpm per SG.

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO
(continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of a charging pump operating at its low speed setting. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, LEAKAGE through two in-series PIVs, and primary to secondary LEAKAGE, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal return (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Each Steam Generator (SG)

Total primary to secondary LEAKAGE amounting to 0.1 gpm through each SG produces acceptable offsite doses and tube stresses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident or result in a coincident SGTR. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE. The SGs shall also be OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the in-series PIVs in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any RCS pressure boundary LEAKAGE exists, or if the Required Action of Condition A cannot be completed within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

(continued)

D
BASES

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE which is not allowed by this LCO, would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the RCS at steady state operating conditions. Therefore, this SR is required to be performed once during the initial 12 hours of steady state operation and every 72 hours thereafter.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and volume control tank levels, makeup and letdown, and RCP seal injection and return flows.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1 (continued)

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16, Issued for comment July 10, 1967.
 2. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
 3. UFSAR, Section 15.6.3.
 4. Letter from R. A Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and Atomic Industry Forum (AIF) GDC 53 (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in-series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both in-series PIVs for a given line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through in-series valves is determined by a water inventory balance (SR 3.4.13.1) or other confirmatory tests. A known component of the identified LEAKAGE before operation begins is the least of the individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight. Prior to the required surveillance testing (SR 3.4.14.1) and water inventory balance (SR 3.4.13.1) in MODES 3 and 4, any leakage through the PIVs is considered unidentified LEAKAGE.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment (i.e., intersystem LOCA), an unanalyzed accident, that could degrade the ability for low pressure injection.

(continued)



BASES

BACKGROUND
(continued)

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core damage. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs and to identify which configurations dominate the risk profile for intersystem LOCA potential. In response to Reference 6, a plant specific evaluation of intersystem LOCAs was performed to identify the most risk significant configurations.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE
SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core damage. The dominant accident sequence in the intersystem LOCA category as identified by Reference 4 was the failure of the low pressure portion of the RHR System outside of containment. This accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent increased risk of core damage.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. In response to Reference 6, a plant specific evaluation of intersystem LOCAs was performed. PIVs in the following systems connected to the RCS were evaluated:

- a. residual heat removal (RHR);
- b. safety injection (SI); and
- c. chemical and volume control.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The evaluation of intersystem LOCAs concluded that several configurations identified in References 4 and 5 existed in the RHR and SI systems. The PIV configurations in the Chemical and Volume Control System were not identified as being risk significant due to the installed orifices in the letdown piping and the use of piping designed to RCS pressure conditions from the discharge of the positive displacement pumps to containment (Ref. 7).

The PIVs identified in the SI and RHR Systems are listed below:

- 853A RHR Inlet Check Valve to Reactor Vessel Core Deluge
- 853B RHR Inlet Check Valve to Reactor Vessel Core Deluge
- 867A SI Pump Discharge and Accumulator A Check Valve to RCS Cold Leg B
- 867B SI Pump Discharge and Accumulator B Check Valve to RCS Cold Leg A
- 877A SI Pump Discharge Check Valve to RCS Hot Leg B
- 877B SI Pump Discharge Check Valve to RCS Hot Leg A
- 878A SI Pump Discharge Isolation MOV to RCS Hot Leg B
- 878C SI Pump Discharge Isolation MOV to RCS Hot Leg A
- 878F SI Pump Discharge Check Valve to RCS Hot Leg B
- 878G SI Pump Discharge Check Valve to RCS Cold Leg B
- 878H SI Pump Discharge Check Valve to RCS Hot Leg A
- 878J SI Pump Discharge Check Valve to RCS Cold Leg A

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. This LCO only applies to those PIVs which are determined to be in the most risk significant configurations (Ref. 7) as listed in Applicable Safety Analysis. The remaining PIVs are governed by LCO 3.4.13, "RCS Operational LEAKAGE" and LCO 3.6.3, "Containment Isolation Boundaries."

(continued)

BASES

LCO
(continued)

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. A leakage rate limit based on valve size is used since this is superior to a single allowable value (Ref. 8).

Reference 9 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

In MODES 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and isolation failures are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

(continued)

BASES

ACTIONS
(continued)

A.1 and A.2

A leaking flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that isolation of the affected flow path with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation. The use of a valve other than the previously leaking PIV must include consideration that the plant may no longer be in an analyzed condition. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage due to reduced RCS pressure while reducing the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1 and SR 3.4.14.2

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve and should be based on an RCS pressure of ± 20 psig of normal system operating pressure. Leakage testing requires a stable pressure condition.

For multiple in-series PIVs, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other in-series valve meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing of the check valves (877A, 877B, 878F, and 878H) and the motor operated valves (878A and 878C) identified as PIVs in the SI hot leg injection lines is to be performed at least once every 40 months. This surveillance interval is allowed since the two SI hot leg injection lines are maintained closed to address pressurized thermal shock (PTS) concerns. Each injection line is isolated by two check valves and one motor operated valve in-series which must all fail to create the potential for an intersystem LOCA. Testing of the remaining RCS PIVs in the SI and RHR systems is to be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 10) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

D
BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1 and SR 3.4.14.2 (continued)

In addition to the periodic testing requirements, testing must be performed once after the valve has been opened by flow, exercised, or had maintenance performed on it to ensure tight reseating. This maintenance does not include minor activities such as packing adjustments which do not affect the leak tightness of the valve. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been resealed. A limit of 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. Atomic Industry Forum (AIF) GDC 53, Issued for comment July 10, 1967.
4. WASH-1400 (NUREG-75/014), "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, October 1975.
5. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," May 1980.
6. Generic Letter, "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980.

(continued)

BASES

REFERENCES
(continued)

7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.
 8. EG&G Report, EGG-NTAP-6175.
 9. ASME, Boiler and Pressure Vessel Code, Section XI.
 10. 10 CFR 50.55a(g).
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

Atomic Industry Forum (AIF) GDC 16 (Ref. 1) requires that means be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (RCPB). AIF-GDC 34 (Ref. 1) also requires that the RCPB be designed to reduce the probability of rapid propagation failures. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. The leakage detection systems support these requirements by both detecting RCS LEAKAGE and identifying the location of its source.

Industry practice has shown that small water flow changes can be readily detected in contained volumes by monitoring changes in water level or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE (i.e., containment sump A) is monitored for level and sump pump actuation and can measure approximately a 2.0 gpm leak in one hour. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. The particulate monitor (R-11) can detect a leak of 0.013 gpm within 20 minutes assuming the presence of corrosion products. The gaseous monitor (R-12) can detect a leak of 2.0 to 10.0 gpm within 1 hour and is considered a backup to the particulate monitor. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

(continued)

BASES

BACKGROUND
(continued)

Alternative means also exist to monitor RCS LEAKAGE inside containment. These include humidity detectors, air temperature and pressure monitoring, and condensate flow rate from the air coolers. The capability of these systems to detect RCS leakage is influenced by several factors including containment free volume and detector location. These systems are most useful as alarms or indirect indicating devices available to the operators and are not required by this LCO (Ref. 2).

The leakage detection systems are also used to support identification of leakage from open systems found in containment. This includes service water and fire service water systems. Leakage from these systems is required to be monitored in response to IE Bulletin No. 80-24 (Ref. 3).

APPLICABLE
SAFETY ANALYSES

The asymmetric loads produced by the postulated breaks are the result of an assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that cause large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These asymmetric loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The resolution of USI-2 for Westinghouse PWRs was use of fracture mechanics technology for RCS piping > 10 inches diameter (Ref. 5). This technology became known as leak-before-break (LBB). Included within the LBB methodology was the requirement to have leakage detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of LBB for Ginna Station is documented in Reference 6.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the plant and the public. Required corrective actions are provided in LCO 3.4.13, RCS Operational LEAKAGE. The capability of the leakage detection systems was evaluated by the NRC in Reference 7.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

(continued)

BASES

LCO
(continued)

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump A monitor (level or pump actuation from either sump A pump), in combination with a gaseous (R-12) or particulate (R-11) radioactivity monitor provides an acceptable minimum. Alternatively, the plant vent gaseous (R-14) or particulate (R-13) monitors may be used in place of R-12 and R-11, respectively, provided that a flowpath through normally closed valve 1590 is available and R-14A is OPERABLE.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1.1, A.1.2, and A.2

With the required containment sump A monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. In addition to an OPERABLE gaseous or particulate atmosphere monitor, the containment air cooler condensate collection system must be verified to be OPERABLE within 24 hours, or the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. The use of the gaseous monitor (R-12) is acceptable due to the increased frequency of performing SR 3.4.13.1 or the use of the containment air cooler condensate collection system.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

The containment air cooler condensate collection system is OPERABLE if the flow paths from all four containment air coolers to their respective collection tanks are available and a CHANNEL CALIBRATION of the monitor has been performed within the last 24 months. The containment air cooler condensate collection system is provided as an option for detecting RCS leakage since SR 3.4.13.1 is not performed until after 12 hours of steady state operation. Therefore, this collection system can be used during MODE changes if the containment sump monitor is inoperable.

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Actions A.1.1, A.1.2, and A.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1.1, B.1.2, and B.2.1

With both gaseous (R-12) and particulate (R-11) containment atmosphere radioactivity monitoring instrumentation channels inoperable (and their alternatives R-13 and R-14), alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a grab sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes that at least one other form of leakage detection is available.

(continued)

BASES

ACTIONS

B.1.1, B.1.2, and B.2.1 (continued)

Required Actions B.1.1, B.1.2, and B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitors are inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1.1, C.1.2, C.2.1, and C.2.2

With the required containment sump monitor and the particulate containment atmosphere radioactivity monitor (R-11) inoperable, the only installed means of detecting leakage is the gaseous containment atmosphere radioactivity monitor (R-12). This condition does not provide a diverse means of leakage detection. Also, the gaseous monitor can only measure between a 2.0 and 10.0 gpm leak within 1 hour which may not meet the 1.0 gpm in less than four hours detection rate required by Generic Letter 84-04 (Ref. 5).

The Required Actions are to analyze grab samples of the containment atmosphere or perform RCS water inventory balance, SR 3.4.13.1, at a frequency of 24 hours. The combination of the gaseous monitor and either the periodic grab samples or RCS inventory balance provide information that is adequate to detect leakage. Restoration of either of the inoperable monitors to OPERABLE status within 30 days is required to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period of time.

Required Actions C.1.1, C.1.2, C.2.1, and C.2.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor and particulate containment atmosphere radioactivity monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If a Required Action of Condition A, B, or C cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

This SR requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

This SR requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months considers channel reliability and operating experience has proven that this Frequency is acceptable.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16 and 34, Issued for comment July 10, 1967.
 2. Regulatory Guide 1.45.
 3. IE Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment."
 4. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.
 5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
 6. Letter from D. C. DiIanni, NRC, to R. W. Kober, RG&E, Subject: "Generic Letter 84-04," dated September 9, 1985.
 7. NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Nuclear Power Plant," December 1982.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity are provided in the SRs. DOSE EQUIVALENT I-131 is calculated using Table E-7 of Regulatory Guide 1.109 (Ref. 2). The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE
SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 0.5 gpm.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the plant that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity (Ref. 4). One case assumes specific activity at $1.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 500 for a duration of four hours immediately after the accident. The second case assumes the initial reactor coolant iodine activity at $60.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/E \mu\text{Ci/gm}$ for gross specific activity.

The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. The analysis also assumes a loss of offsite power at the same time as the reactor trip following the SGTR event.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric relief valves and the main steam safety valves. This steam release continues for eight hours until the residual heat removal system is utilized for cooldown purposes. All noble gas activity in the RCS which is transported to the secondary system by the tube rupture is assumed to be immediately released to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the initial cooldown ends and the RCS system pressure stabilizes below the relief valve setpoint.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1 for more than 7 days.

The increased permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 7 day time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but they would still be within 10 CFR 100 dose guideline limits.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to $1.0 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$, and the gross specific activity in the reactor coolant is limited to $100/E \mu\text{Ci/gm}$ (where E is the average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

(continued)

BASES (continued)

APPLICABILITY In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 8 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 8 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 7 days if the limit violation resulted from normal iodine spiking.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the DOSE EQUIVALENT I-131 is greater than the LCO limit and within the acceptable range of Figure 3.4.16-1. This allowance is provided because of the significant conservatism included in the LCO limit. Also, reducing the DOSE EQUIVALENT I-131 to within limits is accomplished through use of the Chemical and Volume Control System (CVCS) demineralizers. This cleanup operation parallels plant restart following a reactor trip which frequently results in iodine spikes due to the large step decrease in reactor power level and RCS pressure excursion. The cleanup operation can normally be accomplished within the LCO Completion Time of 7 days.

(continued)

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1951

BASES

ACTIONS
(continued)

B.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 specific activity is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 8 hours. The change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event. The Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If the gross specific activity is not within limit, the change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1

This SR requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1 (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg} \geq 500^{\circ}F$. The 7 day Frequency considers the unlikelihood of a gross fuel failure during this time.

SR 3.4.16.2

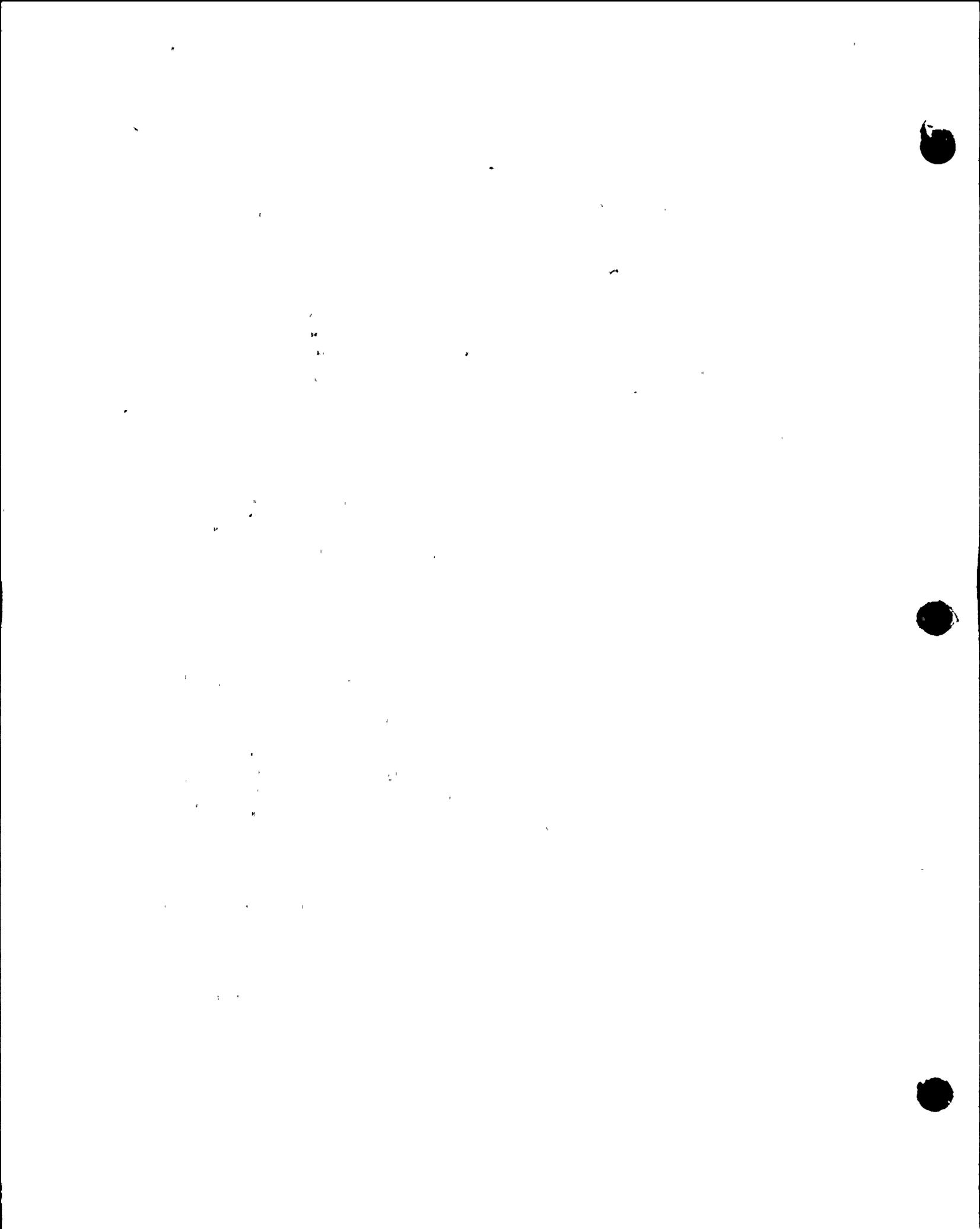
This SR is only performed in MODE 1 to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more likely to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 10 hours after a power change $\geq 15\%$ RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for \bar{E} determination is required within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours and every 184 days (6 months) thereafter. This ensures that the radioactive materials are at equilibrium so the analysis for \bar{E} is representative and not skewed by a crud burst or other similar abnormal event. The \bar{E} determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for \bar{E} is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency recognizes \bar{E} does not change rapidly.

This SR is modified by a Note that indicates sampling is only required to be performed in MODE 1 such that equilibrium conditions are present during the sample.

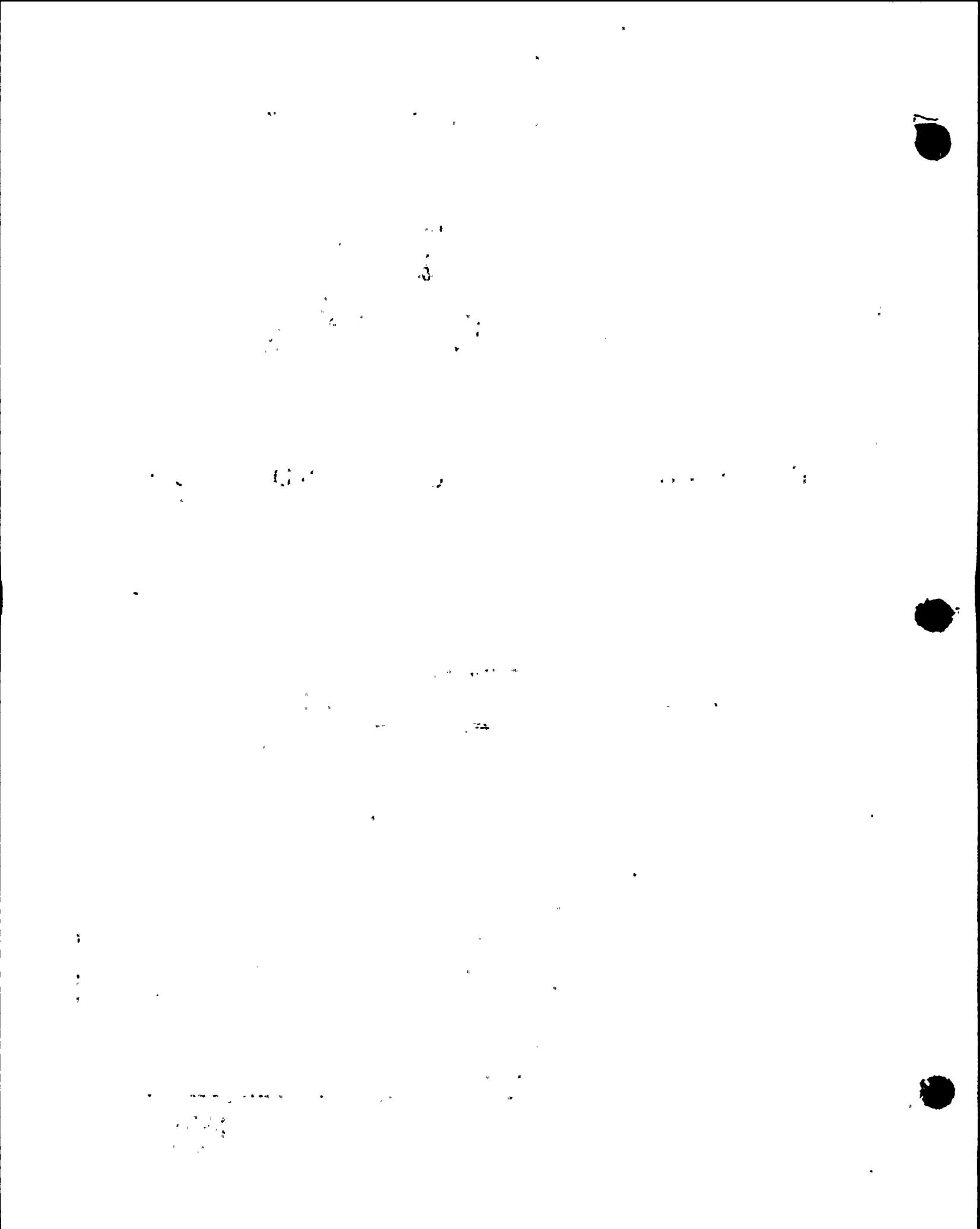
(continued)



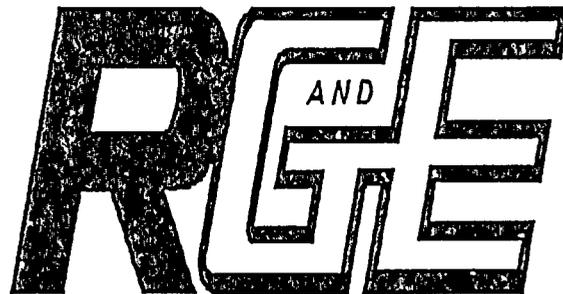
BASES (continued)

REFERENCES

1. 10 CFR 100.11.
 2. Regulatory Guide 1.109, Revision 1.
 3. UFSAR, Section 15.6.3.
 4. WCAP-11668, "LOFTTR2 Analysis of Potential Radiological Consequences Following a SGTR at the R.E. Ginna Nuclear Power Plant," November 1987.
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12/28/95
9601030010



Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

Attachments I, J, K

Volume IV

ATTACHMENT I

"Redlined" Version of Attachment A, Sections D, E, F, and G
as Submitted on May 26, 1995

December 1995

D. JUSTIFICATION (CURRENT GINNA TS)

Converting to the ITS format will provide a significant human factors improvement by locating similar requirements within the same section and also provide a standard structure. In addition, the expanded bases information will support preparation of safety evaluations and training activities. There are several types of changes that are being requested by this LAR in order to perform the conversion. These changes are with respect to both the ITS and the current Ginna Station Technical Specifications. The technical and significant administrative changes related to the current Ginna Station TS are organized into multiple categories as summarized below.

i. Relocation of Requirements Within Technical Specifications

Many current specifications are moved to support consolidation of similar requirements within the same section. Since the requirements are only being relocated within the technical specifications, there is no reduction in safety. This category is mainly used to identify multiple requirements that are consolidated into a single new specification and not for listing requirements which are only renumbered.

ii. Elimination of Duplicated Regulatory Requirements

Several specifications currently duplicate existing regulatory requirements. The removal of these specifications eliminates the need to change technical specifications when there are rule changes. Since all licensees must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicated requirements from technical specifications. The implementation of these requirements are contained in procedures and other licensee controlled documents.

iii. Relocation of Current Requirements To Other-Controlled Documents

The relocation of certain requirements to other licensee controlled documents (i.e., UFSAR, QA Program, and plant procedures) does not eliminate the requirement. Instead, the requirements are relocated to other more appropriate documents and programs which have sufficient controls in place to manage implementation and future changes (e.g., 10 CFR 50.54(a)(3) and 10 CFR 50.59). The relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety.

iv. Addition of New ITS Requirements

There are several requirements contained in NUREG-1431 which are not currently in the Ginna Station Technical Specifications. These ITS requirements were added in order to provide a more complete specification. Changes within this category are further identified as either being a "more restrictive change" (iv.a) or a "less restrictive change" (iv.b).

v. Other Changes to Technical Specifications (Technical)

Several changes to existing requirements were made to provide consistency with NUREG-1431. Examples include moving requirements to LCO Notes and revising the current specified Completion Time. Also included within this category are the revision of the existing bases to reflect more current information. Changes within this category are further identified as either being a "more restrictive change" (v.a), "less restrictive change" (v.b), or an "administrative change" (v.c).

vi. Other Changes to Technical Specifications (Administrative)

Several minor changes to the technical specifications were made that are minor revisions only and do not involve any technical issues. Examples include updates of references to the Code of Federal Regulations.

The following section discusses changes to the current Ginna Station Technical Specifications which were not addressed in Section C of this attachment. This section is organized based on the existing TS chapter numbers to facilitate easier review. Each change is also identified with respect to one of the above categories (e.g., Ginna Station TS Category (i)). A marked up copy of the Ginna Station Technical Specifications is provided in Attachment B which identifies major changes only. A cross reference is provided in the margin of each specification that has been changed by use of a circle containing section numbers from below. For example, "1.i" found in the margin of the markup would refer to section 1.i below. A cross reference between the ITS and current Ginna Station Technical Specifications is also provided in Attachment E.

1. Technical Specification 1.0

- i. TS 1.2 - The definitions of operating MODES were revised as follows (these are Ginna TS Category (v.a) changes):
 - a. Refueling - see Note 1.ii below.
 - b. Cold Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits.

- c. Hot Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits. The average reactor coolant temperature was also revised from $\geq 540^\circ\text{F}$ to $\geq 350^\circ\text{F}$. This change eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2. The expansion of this temperature range is conservative since the current TS only use the Hot Shutdown MODE in two aspects. The first method is requiring a shutdown to this mode due to plant conditions. Since the upper temperature range for Hot Shutdown remains the same (i.e., the Operating MODE temperature), there is no impact. The second method is to require certain equipment to be OPERABLE in this mode. However, lowering the temperature limit to 350°F requires that the equipment would be OPERABLE for a larger temperature range.
- d. Operating - The reactivity limit was revised from $> -1 \Delta k/k\%$ to $\geq 0.99 k_{off}$ which are equivalent limits. The average reactor coolant temperature of $\sim 580^\circ\text{F}$ was not added since this parameter is specified in new LCO 3.4.1. In addition, the Operating MODE was separated into two modes: Operating and Startup. The only difference between these two modes is that Startup is defined when the reactor is $\leq 5\%$ Rated Thermal Power (RTP) while the Operating MODE is when the reactor is $> 5\%$ RTP.
- e. A new operating mode (Hot Standby) was provided between Hot Shutdown and Cold Shutdown. This mode is defined as when the reactivity condition is $< 0.99 k_{eff}$ and the average reactor coolant temperature is $< 350^\circ\text{F}$ and $> 200^\circ\text{F}$ when the reactor vessel head closure bolts are fully tensioned. The definition of this new mode eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2.

- ii. TS 1.3 - This definition of refueling was deleted. The current TS 1.2 provides a definition of refueling as being the reactor mode when reactivity is $\leq -5 \Delta k/k\%$ and the average reactor coolant temperature is $\leq 140^\circ\text{F}$. TS 1.3 states that refueling is "any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted" which is a subset of the mode defined in TS 1.2. The new TS Table 1.1-1 states that refueling is any condition in which "one or more reactor vessel head closure bolt is less than fully tensioned" with fuel in the reactor. While an average reactor coolant temperature or reactivity limit is no longer provided for the refueling mode definition, the reactor vessel head closure bolts cannot be removed at elevated reactor coolant temperatures or when the RCS is pressurized due to their design. A reactivity limit is also not required when the RCS is depressurized. Therefore, the new definition of the refueling mode is more conservative than current TS 1.3 and generally consistent with TS 1.2. This is a Ginna TS Category (v.a) change.
- iii. TS 1.5 - The definition for Operating was not added to the new specifications since it is no longer required. This definition is addressed by the new definition for OPERABLE - OPERABILITY. This is a Ginna TS Category (i) change.
- iv. TS 1.6 - The definition for Degree of Redundancy (Instrumentation Channels) was not added to the new specifications since it is no longer required. This definition is addressed within new TS 3.3 (Instrumentation). This is a Ginna TS Category (v.c) change.
- v. TS 1.7.1 - This was revised to specify that the CHANNEL CALIBRATION includes the required interlock and time constant functions of the channel. In addition, discussion of calibrating instrument channels with resistance temperature detectors was added for clarification. These are Ginna TS Category (v.a) changes.
- vi. TS 1.7.2 - The last sentence of this definition was revised as follows:

No change

This determination shall include, where possible, comparison of the channel indication and/or status with to other indications and/or or status derived from independent instrumentation channels measuring the same parameter.

These minor changes provide greater clarification of the defined term and are Ginna TS Category (v.c) changes.

- vii. TS 1.7.3 - The definitions for testing of analog and bistable channels were combined into one description with a new title. The only difference between the two definitions is that testing of bistable channels required injection of a simulated or source signal into the sensor versus "as close to the sensor as possible" for analog channels. Since the bistable must be actuated to determine operability, maintaining the analog channel description for the combined definition is acceptable. In addition, the combined definition was expanded to require "adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy." These are Ginna TS Category (v.a) changes.
- viii. TS 1.7.4 - The definition for Source Check was not added to the new specifications since it is no longer required. The performance of a Source Check is now addressed within the definition of CHANNEL CALIBRATION and CHANNEL OPERATING TEST (COT). This is a Ginna TS Category (v.c) change.
- ix. 229 TS 1.8 - The definition for Containment Integrity was ~~not added, relocated to the bases of new specifications since it is no longer required~~ TS 3.6.1 and 3.6.2 which essentially requires compliance with 10 CFR 50, Appendix J and the GDC. Containment Integrity ~~is addressed by new~~ Ginna TS 3.6 which essentially requires compliance with 10 CFR 50, Appendix J ~~Category (iii) change. This is a Ginna TS Category (v.c) change.~~
- x. TS 1.10 - The definition for Hot Channel Factors was not added to the new specifications since it is no longer required. The Hot Channel Factor limit is only discussed in one LCO with the limit defined in the COLR. This is a Ginna TS Category (v.c) change.
- xi. TS 1.11 - This previously deleted definition was not added to the new specifications. This is a Ginna TS Category (vi) change.
- xii. TS 1.12 - The Frequency for Surveillance Requirements is now specified in hours, days or months in the new specifications such that the current definition of Frequency Notation is no longer required. Consequently, this definition was replaced with a general description of how to use and apply the Frequency requirements. In addition, the definition of refueling Frequency was revised from 18 months to 24 months for all systems. This is discussed in Attachment H and is a Ginna TS Category (v.b.1) change.
- xiii. 157 TS 1.13 - The definition for Offsite Dose Calculation Manual (ODCM) was ~~not added, moved to the new specifications since it is no longer required~~ ODCM program description in



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ITS specification 5.5.1. The ODCM change to the CTS is described in new Specification 5.5.1 editorial because the program description involves reorganization or reformatting of requirements without affecting technical content. This is a Ginna TS Category (v.c) change.

xiv. TS 1.14 - The definition for Process Control Program (PCP) was not added to the new specifications since it is no longer required. The PCP was relocated from the technical specifications to the TRM and does not need to be described within new TS 1.1. This is a Ginna TS Category (v.c) change.

xv. TS 1.15 - The definition for Solidification was not added to the new specifications since it is no longer required. Solidification is described within the PCP which was relocated from the technical specifications to the TRM. Therefore, this definition does not need to be provided in new TS 1.1. This is a Ginna TS Category (v.c) change.

xvi. TS 1.16 - The definition for Purge - Purging was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3. This is a Ginna TS Category (v.c) change.

xvii. TS 1.17 - The definition for Venting was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3. This is a Ginna TS Category (v.c) change.

xviii. ~~TS 1.18 - The reference to the "dose conversion factors for adult thyroid dose via inhalation" was not added to the new specifications since a specific reference to Table E 7 of Regulatory Guide 1.109 was added. This table only contains dose conversion factors for adults via inhalation. Therefore, the existing reference is no longer necessary. This change is consistent with Traveller WTS 1, C.2. This is a Ginna TS Category (vi) change.~~

xix. TS 1.19 - The definition for Reportable Event was not added to the new specifications since it is no longer required. Reportable Events are described in 10 CFR 50.72 and 50.73. This is a Ginna TS Category (ii) change.

xx. TS 1.20 - The definition for Canisters Containing Consolidated Fuel Rods was not added to the new specifications since it is no longer required. This definition is provided in new TS 4.3 which is the only section that addresses consolidated fuel rods. This is a Ginna TS Category (v.c) change.

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- xxi. TS 1.21 - The definition for Shutdown Margin was expanded to require another assumption that in MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature. Also, the definition was revised to require consideration of any RCCA known to be incapable of being fully inserted. This is in addition to the existing assumptions related to a stuck fully withdrawn single RCCA with the highest reactivity worth. The definition description discussing "no changes in xenon or boron concentration" was deleted since this level of detail is not required. These clarifications, which are consistent with NUREG-1431, are Ginna TS Category (v.a) changes.
- xxii. TS 1.4 - The definition for OPERABLE - OPERABILITY was revised to remove "supports." This phrase was added to the current definition by Reference 3 but is not consistent with the definition as provided in NUREG-1431. Therefore, to provided consistency, this was not added to the new specifications. This is a Ginna TS Category (v.c) change.
- xxiii. The following definitions were added to the new specifications since the associated terms are used throughout the document (these are Ginna TS Category (v.a) changes):
- a. ACTIONS
 - b. ACTUATION LOGIC TEST
 - c. AXIAL FLUX DIFFERENCE
 - d. CORE ALTERATION
 - e. CORE OPERATING LIMITS REPORT (COLR)
 - f. LEAKAGE
 - g. PHYSICS TESTS
 - h. PRESSURE TEMPERATURE LIMITS REPORT (PTLR)
 - i. RATED THERMAL POWER
 - j. STAGGERED TEST BASIS
 - k. TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)
- xxiv. A new section was added to the specifications which explains the use of Logical Connectors within the new TS. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.
- xxv. A new section was added to the specifications which explains the use of the Completion Time convention within the new TS. There are several changes from the current Ginna Station TS format which are discussed in this section (these are Ginna TS Category (v.a) changes):

- a. Completion Times in the new TS are based on the format that the clock for all Required Actions begin from the time that the Condition is entered. The Completion Times in the new specifications and the current Ginna Station TS are typically equal. For example, the new specifications may require that the plant be in MODE 3 within 6 hours and in MODE 4 within 36 hours for a specified Condition while the current Ginna Station TS require that the plant be in MODE 3 within 6 hours and in MODE 4 within an additional 30 hours for the same Condition. The intent of both the new specifications and the current Ginna Station TS is the same (i.e. be in MODE 4 within 36 hours).
- b. The new specifications restrict multiple entries into the ACTION table for separate Conditions unless it is specifically stated as acceptable. For example, if one SI pump is inoperable and during the LCO, a second SI pump is declared inoperable, the plant would enter 3.0 conditions in both the new specifications and the current Ginna Station TS. If the first SI pump were restored to OPERABLE status before entering MODE 3, the plant could resume operation in both TS. However, in the current TS, the Completion Time for restoring the second SI pump to OPERABLE status would begin from the time that it was declared inoperable. In the new specifications, the Completion Time would begin from the time the first pump was declared inoperable with an additional 24 hours allowed. This is a conservative change.

xxvi. A new section was added to the specifications which explains the use of the Frequencies specified within the SRs. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.

2. Technical Specification 2.1

- i. The Applicability was revised to ~~not only include~~ define when the reactor is in "operation" or critical, but also when in MODE 2 as MODES 1 and subcritical². This ensures that the Reactor Core Safety Limits are also met during reactor startup since there is a potential for an inadvertent criticality with the reactor near normal operating temperature editorial change only since "operation" has been redefined as MODES 1 and pressure conditions² per Section D Change 1.j.d. This is a Ginna TS Category (iv.a) change.

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3. Technical Specification 2.2

- i. The Applicability was revised to "MODES 1, 2, 3, 4, and 5." The proposed Applicability does not require this Safety Limit (SL) to be met when fuel is in the vessel with one or more reactor vessel head closure bolts less than fully tensioned or with the head removed. With the reactor head bolts less than fully tensioned, it is highly unlikely that the RCS can be pressurized greater than the SL pressure due to the low temperature over-pressure protection requirements. With the head removed, it is not possible to pressurize the RCS greater than the SL pressure. This is a Ginna TS Category (v.b.2) change.

4. Technical Specification 2.3.

- i. This entire section was relocated to ITS Chapter 3.3, "Instrumentation." This is a Ginna TS Category (i) change.
- ii. TS 2.3 - Various limiting safety system settings (LSSS) are addressed as "Trip Setpoints," "Allowable Values," or "Applicable Modes" (as permissives) for their respective Reactor Trip System (RTS) instrumentation Functions in new LCO 3.3.1. Specific changes to the LSSS are discussed below for each of the associated Functional Units. This is a Ginna TS Category (i) change.
- iii. ~~TS 2.3.1.2 Not used, d and TS 2.3.1.2.e - Various parameters used in the methodology for determining the Overtemperature ΔT and the Overpower ΔT Functions were not added to the specifications. These parameters are associated with variables which may change as a result of a reload analysis and are relocated to the COLR. This is a Ginna TS Category (iii) change.~~
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- iv. TS 2.3.3.1, TS 2.3.3.2, and Figure 2.3-1 - The LSSS for the loss of voltage and degraded voltage functions were revised to provide a minimum Trip Setpoint value. Criteria for the establishment of equivalent values based on measured voltage versus relay operating time was relocated to the bases for LCO 3.3.4. This is a Ginna TS Category (iii) change.
- v. ~~TS 2.3.1.2 3.2 - The listing of permissives was revised to provide requirements and setpoints for P-6, P-9, and P-10. g - The LSSS These permissives also provide enabling and blocking features for the RCP underfrequency functions was not added to the new specifications various RTS functions. This is justified in Reference 44 which shows that this trip function, though installed at a Ginna Station, is not required or applicable based on the offsite power source design TS Category (v. - This setpoint and requirement are relocated to the TRMa) change. - This is a Ginna TS Category (iii) change.~~
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5. Technical Specification 3.0

- i. A new section LCO 3.0.1 was added which explains the use of the Applicability statement in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.1. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- ii. A new section LCO 3.0.2 was added which explains the use of the associated ACTIONS upon discovery of a failure to meet an LCO in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.2. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- iii. TS 3.0.1 - This was revised to clarify the use of the actions that must be implemented when an LCO is not met and (1) an associated Required Action and Completion Time is not met and no other Condition applies, or (2) the condition of the plant is not specifically addressed by the associated ACTIONS. The current requirement that the LCO time limits apply if they are more limiting than those required by LCO 3.0.3 is deleted and an expanded discussion is provided in the Basis to clarify the applicability of this requirement. This section does not provide any new requirements ~~except as discussed in item 5.viii below~~. The clarifications and examples are based on the use of the new ITS format. This is a Ginna TS Category (v.c) change.
- iv. A new section LCO 3.0.4 was added which explains the limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met in the new TS. This section provides new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.

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- v. A new section LCO 3.0.5 was added to provide an exception to LCO 3.0.2 for instances where restoration of inoperable equipment to an OPERABLE status could not be performed while continuing to comply with Required Actions. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as: maintaining an isolation valve closed or tripping an inoperable instrument channel. To allow the performance of SRs to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the present Specifications. Without this allowance certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications to preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. Since this requirement is informally utilized and has no licensing basis, this section is considered to provide new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.
- vi. TS 3.0.2 - This was deleted and replaced by LCO 3.0.6 which provides guidance regarding the appropriate ACTIONS to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). Since its function is to clarify existing ambiguities and to maintain actions within the realm of previous industry interpretations and NRC positions, this new provision does not provide any new requirements. The information contained in TS 3.0.2 was relocated to LCO 3.8.1 which allows one power source to a safeguards bus and a redundant safety features on a second bus to be inoperable for 12 hours versus 1 hour. This change is consistent with NUREG-1431. These are Ginna TS Category (v.c) and (i) changes, respectively.

vii. A new section LCO 3.0.7 was added to provide guidance regarding Test Exceptions for LCO 3.1.8. This LCO allows specified Technical Specification requirements to be changed (i.e., made applicable in part or whole, or suspended) to permit the performance of special tests or operations which otherwise could not be performed. If this Test Exception LCO did not exist, many of the special tests and operations necessary to demonstrate select plant performance characteristics, special maintenance activities and special evolutions could not be performed. This Specification eliminates the confusion which would otherwise exist as to which LCOs apply during the performance of a special test or operation. Without this specific allowance to change the requirements of another LCO, a conflict of requirements could be incorrectly interpreted to exist. This section does not provide any new requirements. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.

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TS 3.0.1 - This was revised to remove the 1 hour allowance to prepare for a plant shutdown. Instead, the plant must now be in hot shutdown (i.e., MODE 3) within 6 hours of entering this LCO and cold shutdown (i.e., MODE 5) within 36 hours. No time limits are now placed on initiating the plant shutdown, only in the time frame in which the shutdown must be completed. Since the plant must now be in a lower mode in less amount of time, this is a more restrictive change. However, since no restrictions are made as to when the shutdown must commence, this is identified as a Ginna TS Category (v.c) change.

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6. Technical Specification 3.1.1

i. TS 3.1.1.1.b - This requirement was changed to require entry into MODE 1 \leq 8.5% RTP within ~~four~~ six hours versus an immediate power reduction under administrative control. This change defines a specific number of hours to reach this condition which provides greater clarity to the operators. The remaining actions as specified by TS 3.1.1.1.b were relocated to LCO 3.4.5 and are discussed in 6.ii below. This is a Ginna TS Category (v.a) change.

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ii. TS 3.1.1.1.b, 3.1.1.1.c, and 3.1.1.1.d - These requirements were revised per new LCO 3.4.5 to require both reactor coolant loops OPERABLE with one loop in operation during MODES 1 \leq 8.5% RTP, and MODES 2 and 3, versus one in operation and the other OPERABLE for natural circulation between 350°F and 8.5% RTP. However, one RCS loop is now allowed to be inoperable for up to 72 hours provided that the shutdown margin as provided in the COLR is maintained and the non-operating RCS loop is OPERABLE (i.e., available for natural recirculation). These are all conservative changes (Ginna TS Category (iv.a) changes) since:

- a. Two RCS loops are required to be OPERABLE.
- b. A defined period of time is now specified for one RCS loop operation which addresses the concern raised by Reference 12. In addition, Completion Times are now specified for verifying shutdown margin and natural circulation capability.

iii. TS 3.1.1.1.f - The exception for not requiring the RCS or RHR loops during steam generator crevice cleaning operations was not added to the new specifications since RG&E no longer performs this activity and the new SGs scheduled to be installed in 1996 do not have crevices subjected to cleaning as described in this specification. This is a conservative deletion and is a Ginna TS Category (v.a) change.

iv. TS 3.1.1.1.g - The action to be in Cold Shutdown (i.e., < 200°F) within 24 hours was not added for the Condition with both RHR loops inoperable and only one RCS loop inoperable consistent with Condition B of LCO 3.4.6. Since RHR is the only system which provides long-term decay heat removal below 200°F, it is not prudent to bring the plant to a lower MODE until RHR is recovered. This is a Ginna TS Category (v.a) change.

v. TS 3.1.1.1.k - This requirement was changed into a Note for LCO 3.4.6 and 3.4.7. This is a Ginna TS Category (v.c) change.

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The value for LTOP enable temperature with respect to the RCPs was also relocated to the PTLR. This is a Ginna TS Category (iii) change.

- vi. TS 3.1.1.1.f - This requirement was revised to require one RHR loop to be operating when in MODE 5 consistent with LCO 3.4.7 and 3.4.8. A RHR pump is required to be operating since a RCP cannot be routinely operated under these low temperature and pressure conditions. However, a SG with minimum water level of 16% can provide an alternate means of decay heat removal to the operating RHR loop in MODE 5 with the loops filled. In addition, a limit of 15 minutes (versus 1 hour) was placed on removing both RHR loops from service in MODE 5 with the loops not filled due to the reduced RCS inventory. These are conservative changes to the current requirements and are Ginna TS Category (v.a) changes.
- vii. TS 3.1.1.1.e - The note associated with the power sources for the RHR loops has been relocated to the specifications for electrical requirements during MODES 5 and 6 (i.e., LCOs 3.8.2, 3.8.5, 3.8.8, and 3.8.10). This is a Ginna TS Category (i) change.
- viii. TS 3.1.1.1.i and 3.1.1.1.j - These requirements were not added due to the expanded specifications provided in new TS 3.4.4, 3.4.5, 3.4.6, 3.4.7, and 3.4.8. The new specifications ensure that the appropriate RCS or RHR loop is available to provide forced flow for decay heat removal and boron mixing. Therefore, these requirements are no longer necessary. This is a Ginna TS Category (v.c) change.

ix. TS 3.1.1.5.a - The lower limit for pressurizer water level (12%) was not added. This lower limit was related to the previous Safety Injection actuation logic which required a coincident low pressurizer level and low pressurizer pressure trip. This logic was modified as a result of IE Bulletin 79-06A (Ref. 45) to eliminate the coincident low pressurizer level trip (Ref. 46) such that the setpoint is no longer used in an UFSAR Chapter 15 accident analysis. Therefore, the low pressurizer water level setpoint is not required. This is a Ginna TS Category (v.b.3) change.

x. TS 3.1.1.5.b - The current exception for not requiring the pressurizer heaters and water level setpoints during the RCS hydro test was not added to the new specifications. These hydro tests are performed with RCS temperatures below MODE 3 conditions (i.e., < 350°F). Since the new specification only requires the pressurizer to be OPERABLE in MODES 1, 2, and 3, this exception is no longer required. This is a Ginna TS Category (v.a) change.

- xi. TS 3.1.1.6 - The requirement for the reactor vessel head vents was not added to the new specifications since these vents do not meet the criteria specified in the NRC Policy Statement. This is due to the fact that the vents are used to exhaust noncondensable gases and steam from the RCS which could inhibit natural circulation following an accident with an extended loss of offsite power. However, these vents are not the primary success path and are only used by operators if both pressurizer PORVs are unavailable. These vents are not used in the safety analyses nor were identified as being risk significant in the Ginna Station Level 2 PRA (Ref. 47). This requirement will be relocated from TS to the TRM. The remaining requirements contained within this specification relate to the pressurizer PORVs and their associated block valves which are addressed in TS 3.1.1.4. These requirements were revised as discussed in Section D, items 6.xiii and 6.xiv below. This is a Ginna TS Category (iii) change.
- xii. TS 3.1.1.3.a and 3.1.1.3.b - These requirements were not added to the new specifications since the pressurizer safety valves do not provide overpressurization protection during Cold Shutdown and Refueling conditions. This is provided by the low temperature overpressure protection (LTOP) requirement as specified in current TS 3.15 and new LCO 3.4.12. Since the pressurizer safety valves do not perform a safety function during these low MODES of operation, these requirements were not retained. These changes also supersede those proposed in Reference 60. This is a Ginna TS Category (v.b.4) change.

xiii. TS 3.1.1.4.a.i and 3.1.1.6 - These were revised to provide separate Required Actions for the PORVs based on the reason for their inoperability. A PORV which is inoperable for automatic functions but capable of manual actuation must be isolated by its block valve consistent with the current requirement. However, a PORV which is incapable of manual cycling is required to be isolated by its block valve within 1 hour and repaired within 72 hours or the plant must initiate a controlled shutdown. In addition, with both PORVs inoperable, a controlled shutdown to MODE 3 conditions with RCS < 500°F must be accomplished within 8 hours. This limit on operation with an inoperable PORV is provided since a SGTR event cannot be mitigated under this condition. The 72 hours for one inoperable PORV is allowed since the second PORV is available. These changes also supersede those proposed in Reference 60. This is a conservative revision and a Ginna TS Category (iv.a) change.

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- xiv. TS 3.1.1.4.a.ii and 3.1.1.6 - This was revised to require that one or both inoperable block valves valve must be restored to OPERABLE status within 72 hours, both block valves within 7 days, or the plant must initiate a controlled shutdown. This limit on operation with an inoperable block valve is provided since a stuck open PORV cannot be isolated in this condition. These changes also supersede those proposed. The time limits provide adequate time to perform most repairs at power since the valves are located inside containment in Reference 60 the pressurizer cubicle. This is a conservative revision and a Ginna TS Category (iv) These changes also supersede those proposed in Reference 60. This is a conservative revision with respect to current requirements and a Ginna TS Category (iv.a) change.
- xv. TS 3.1.1.2 - This was not added since this temperature limit is not required for safe operation. All necessary heatup and cooldown rates are relocated to the PTLR while new LCO 3.4.1 provides limits on RCS pressure, temperature, and flow. This is a Ginna TS Category (v.b.5) change.
- xvi. TS 3.1.1.3.d - A Note was added which allows the pressurizer safety valves to be removed from service above 350°F for the purpose of setting the valves under hot (i.e., ambient) conditions consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.

xvii. TS 3.1.1.3.c - This was revised to change the pressurizer safety valve lift settings from 2485 psig $\pm 1\%$ to 2485 psig + 2.4%, -3%. The valve lift settings are required to be set to within $\pm 1\%$ following testing; however the OPERABILITY tolerances have been revised. The increased OPERABILITY tolerances have been evaluated in the most limiting pressure transients for Ginna Station (i.e., loss of external load and locked rotor events) and found to result in acceptable results with respect to the safety limit values. This change is a result of an event in which the pressurizer safety valves were found to have drifted outside the existing $\pm 1\%$ tolerance band following testing (Ref. 58). ~~Revising the OPERABILITY tolerances will reduce the potential for future LERs for an issue which has been demonstrated to remain~~ the proposed change is within the accident analysis requirements ASME tolerances of $\pm 1\%$ following testing and $\pm 3\%$ for OPERABILITY. This is a Ginna TS Category (v:b.45) change.

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7. Technical Specification 3.1.2

- i. TS 3.1.2.1.a, Figure 3.1-1, and Figure 3.1-2 - The RCS temperature and pressure curves and the RCS heatup and cooldown curves and limits were relocated from technical specifications to the PTLR which is addressed under Administrative Controls. This is a Ginna TS Category (iii) change.

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- ii. TS 3.1.2.1.b - The requirement for periodically recalculating the RCS temperature and pressure curves and the RCS heatup and cooldown curves and limits was ~~relocated/deleted~~ from technical specifications to the PTLR. A periodic review is already required by 10 CFR 50, Appendix H which does not need to be restated within the technical specifications. This is a Ginna TS Category ~~(iii)~~ ~~(ii)~~ change.
 - iii. TS 3.1.2.1.c.1 - The time allowed to perform an engineering analysis to determine that the RCS is acceptable to continue operation after a pressure and/or temperature limit is exceeded was increased from 6 hours to 72 hours. A duration of 6 hours is not sufficient time to accomplish the required engineering analysis, especially if the event were to occur during evening or early morning hours with limited staff support immediately available. Since NRC accepted guidance for performing the necessary calculations exists, allowing 72 hours to complete the analyses is appropriate, especially since the duration of event is very limited (i.e., controlled by LCO 3.4.3). This is a Ginna TS Category (v.b.6) change.
 - iv. TS 3.1.2.2 - This was not added since this temperature limit is not required for safe operation. All necessary heatup and cooldown rates are relocated to the PTLR while new LCO 3.4.1 provides limits on RCS pressure, temperature, and flow. This is a Ginna TS Category (v.b.5) change.
 - v. TS 3.1.2.3 - This was revised to relocate the pressurizer heatup and cooldown rates to the PTLR. The maximum temperature difference between the pressurizer and spray fluid was not added since this limit is controlled by the cooldown curves. These are Ginna TS Category (iii) and (v.c) changes respectively.

8. Technical Specification 3.1.3

- i. TS 3.1.3.1 - This was revised to raise the minimum temperature for criticality from 500°F to 540°F. This change was made to correct a discrepancy between the definition of reactor operating modes and this requirement. Currently, Ginna Station TS 1.2 defines Hot Shutdown as Reactivity $\leq -1 \Delta k/k\%$ and $T_{avg} \geq 540^\circ F$. In order to achieve criticality at 500°F, the Hot Shutdown condition would have to be directly bypassed. A value of 540°F was selected for the new minimum temperature for criticality based on previous operating experience during startup conditions. This is a Ginna TS Category (v.a) change.

- ii. TS 3.1.3.2 - This was not added since LCO 3.4.2 specifies the minimum temperature for criticality. The minimum temperature with respect to the reactor vessel is contained in the PTLR and is below the limit specified in LCO 3.4.2. This is a Ginna TS Category (v.c) change.
- iii. TS 3.1.3.3 - The existing action statement was revised to require that the plant be in MODE 2 with $k_{\text{eff}} < 1.0$ within 30 minutes if T_{avg} for one or both RCS loops was $< 540^{\circ}\text{F}$ versus subcritical by an amount equal to or greater than the potential reactivity due to depressurization. The new requirement provides clear and precise instructions to operations and ensures that the plant is quickly brought to a condition in which the LCO is no longer applicable. This is a Ginna TS Category (v.c) change.
- iv. TS 3.1.3.1 - The MTC requirements are moved from the RCS chapter in the Ginna Station TS to the Reactivity Control Systems Chapter. This is a Ginna TS Category (i) change.
- v. TS 3.1.3.1 - This was revised to reference cycle specific MTC requirements in the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The MTC maximum upper limit described in TS 3.1.3.1 remains the same in ITS LCO 3.1.4. This is a Ginna TS Category (iii) change.

9. Technical Specification 3.1.4

- i. TS 3.1.4.4 - This specification was revised to only require shutdown to MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 8 hours versus Cold Shutdown within 40 hours consistent with the LCO Applicability. This is a Ginna TS Category (v.c) change.
- ii. TS 3.1.4.1.c - The limit on secondary coolant activity is now required to be met in MODES 1, 2, 3, and 4 and not just when the reactor is critical or RCS temperature is $> 500^{\circ}\text{F}$. The secondary coolant activity limit is based on a steam line break and the resulting dose consequences. A RCS temperature of $> 500^{\circ}\text{F}$ is based on preventing the MSSVs from lifting following a SGTR (i.e., a RCS temperature of $> 500^{\circ}\text{F}$ is only applicable to primary system activity limits not secondary limits). In addition, if the secondary coolant activity limits are not met, TS 3.1.4.4 requires entering cold shutdown (i.e., MODE 5) within 40 hours. Requiring the secondary coolant activity limits to be met for all of MODE 4 (i.e., RCS is $> 200^{\circ}\text{F}$) provides consistency with NUREG-1431 and the current Required Actions if the limit is exceeded. This is a Ginna TS Category (iv.a) change.

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The time to perform a shutdown if secondary activity is not within limits was changed from 8 hours to 6 hours to reach hot shutdown and 32 hours to 30 hours to reach cold shutdown. These completion times are conservative and provide consistency with the rest of the TS. This is a Ginna TS category (v.a) change.

10. Technical Specification 3.1.5

- i. TS 3.1.5.1.1 - Added a new requirement for the containment sump "A" level or pump actuation per LCO 3.4.15. This leakage detection system replaces the containment humidity detectors and the air cooler condensate flow monitor. The containment humidity detectors do not meet the required leakage rate detection capability of 1.0 gpm within 4 hours as required by Generic Letter 84-04 (Ref. 19). In addition, the containment humidity detectors are recommended by RG 1.45 (Ref. 17) to only be used as an alarm or indirect indication of leakage to containment and not as a separate method of detecting leakage. The remaining leakage detection systems provide adequate monitoring as discussed in the new bases and Section C, item 46. These are Ginna TS Category (v.a) changes.

- ii. TS 3.1.5.1.1 and 3.1.5.1.2 - The RCS leakage detection systems are required to be OPERABLE and RCS LEAKAGE within limits above MODE 4 (200°F) and not 350°F per LCO 3.4.15 and 3.4.13. The increased LCO Applicability will address all MODES in which the RCS is at an increased temperature and pressure. This is a Ginna TS Category (iv.a) change.

- iii. TS 3.1.5.1 - Added a note which allows a change in MODE if either the containment sump monitor or both the containment atmospheric radioactivity monitors are inoperable per LCO 3.4.15. This note is appropriate considering the other instrumentation that is available to monitor RCS leakage. This is a Ginna TS Category (v.b.7) change.

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~~iv. TS 3.1.5.2.2.c - The requirement to commence a reactor shutdown with excessive SG tube leakage was revised to allow an additional 4 hours to correct administrative and other similar discrepancies in the Steam Generator Tube Surveillance Program consistent with LCO 3.4.13.B. Requiring a reactor shutdown for most administrative errors is not prudent based on the increased risk for a transient while changing MODES. However, if the integrity of the tube is determined to be inadequate, a reactor shutdown will continue to be immediately initiated. Also, the requirement to perform a SG inspection with excessive leakage if an inspection has not been performed within the last 6 months was not added to the new specifications. Any SG inspections will be determined as part of the corrective actions necessary to repair the leaking tube and in accordance with the Steam Generator Tube Surveillance Program. Since LCO 3.0.4 applies to this LCO, the plant cannot go above MODE 5 without verifying that the SG tube integrity is acceptable. These are Ginna TS Category (v.b.8) changes.~~

11. Technical Specification 3.1.6

- i. TS 3.1.6 - This entire section was not added since RCS Chemistry does not meet the NRC Policy Statement. RCS Chemistry is controlled by plant procedures and is not required to be addressed within the technical specifications. This requirement is being relocated to the TRM. This is a Ginna TS Category (iii) change.

12. Technical Specification 3.2

- i. TS 3.2.5 - The requirement was revised to require placing a charging pump in pull-stop within 1 hour regardless of the status of the RHR pumps or the MODE. This is a conservative change which provides direct operator guidance to perform an action within a defined time period. Also, these requirements were relocated to the LTOP specification to consolidate all related requirements. The verification of the charging pump status every 12 hours was also not added since the plant is required to be in a depressurized and vented condition within 8 hours which removes the need to isolate a charging pump (i.e., a 1.1 square inch vent can mitigate a charging/letdown mismatch event). These are Ginna TS Category (v.a), (i), and (v.c) changes, respectively.
- ii. TS 3.2.1 and TS 3.2.1.1 - The requirements for the boric acid injection flow paths during cold shutdown and refueling which specifies the number of flow paths that must be OPERABLE were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.
- iii. TS 3.2.2 and TS 3.2.4 - The requirements for the boric acid injection flow paths above cold shutdown which specifies the number of flow paths that must be OPERABLE, were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

- iv. TS 3.2.3 and Table 3.2-1 - The requirements for the Boric Acid Storage Tank(s) which specifies the boron concentrations, minimum volume and solution temperature, were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this system do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

13. Technical Specification 3.3

(220) i. TS 3.3.1.1.b and 3.3.1.3 - LCO 3.5.1 Condition A was added, which allows 72 hours to restore accumulator boron concentration to within acceptable limits. The ITS bases state that allowing a longer period of time to correct boron concentration is acceptable since the volume of water in the accumulators is the critical feature. Attempting to correct boron concentration within the current 1 hour limit would create a significant burden on the operations staff. Therefore, the current 1 hour LCO was only maintained for accumulator pressure and volume. In addition, the accumulator boron concentration limits were relocated. Limit was increased to 2100 ppm to support the COLR since these values can change due to value used in the accident analysis for the 18 month refueling cycle changes. These are Ginna TS Category (v.a) upper limit of 2600 ppm was also added to address chemical considerations of the sump fluid following an accident. This value is also consistent with that used for 18 month refueling cycles. 9) and (iii) changes, respectively. These are Ginna TS Category (v.b.9) and (v.a) changes, respectively.

(220) ii. TS 3.3.1.1.a and 3.3.1.2 - LCO 3.5.4.A was added which allows 8 hours to restore the RWST boron concentration to within acceptable limits. The ITS bases state that allowing a longer period of time to correct boron concentration is acceptable since it requires a longer period of time to perform this type of adjustment due to the large volume of water contained within the RWST. In addition, the RWST boron concentration limits were relocated. Limit was increased to 2300 ppm to support the COLR since these values can change due to value used in the accident analysis for the 18 month refueling cycle changes. These are Ginna TS Category (v.a) upper limit of \leq 2600 ppm was also added to address chemical considerations of the sump fluid following an accident. This value is also consistent with that used for 18 month refueling cycles. 10) and (iii) changes, respectively. These are Ginna TS Category (v.

b.10) and (v.a) changes, respectively.

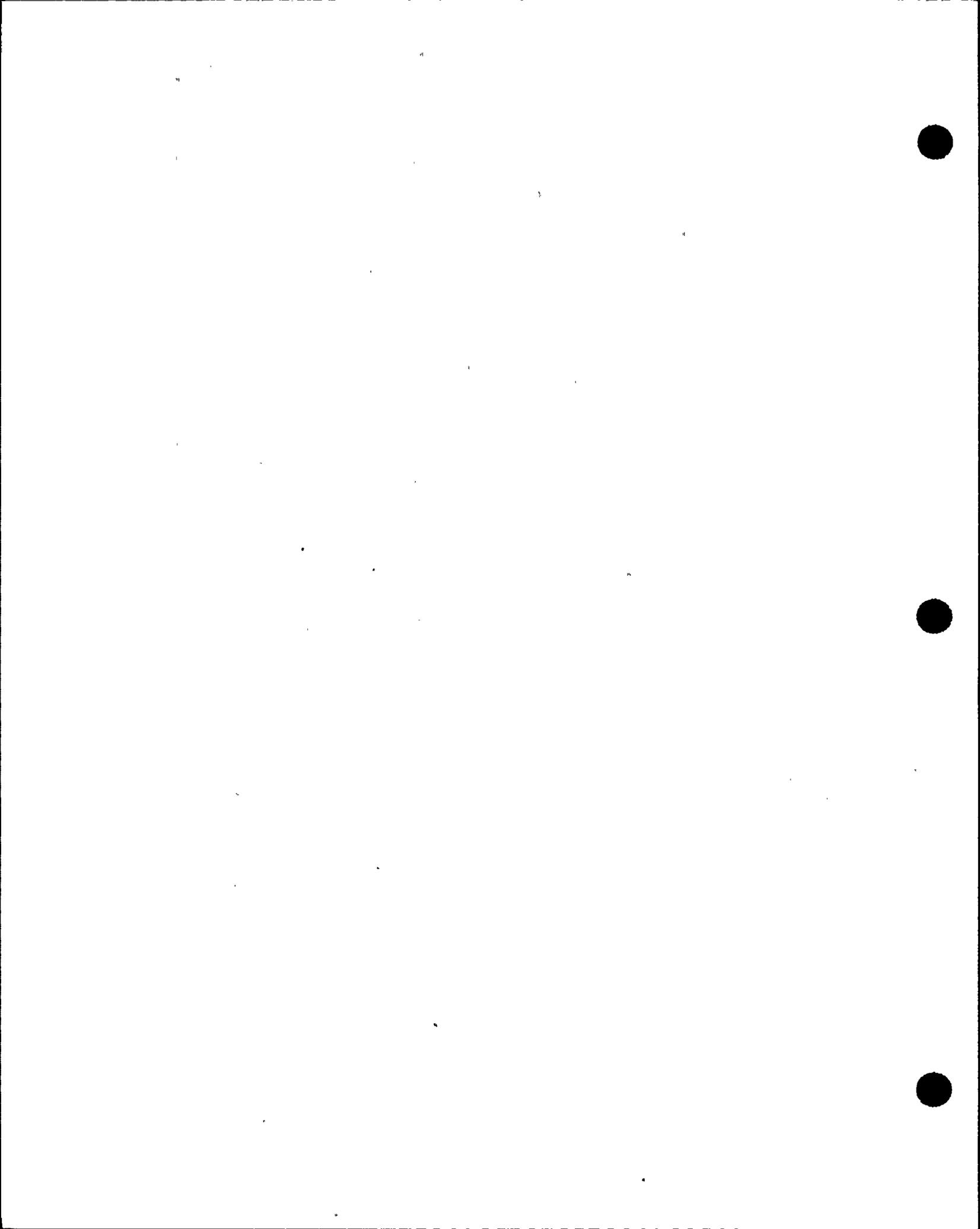
- iii. TS 3.3.1.1.c - Two notes associated with LCO 3.5.2 were added. The first note allows both SI pump flow paths to be isolated for up to 2 hours to perform pressure isolation valve testing. The ITS bases state that this is acceptable since the isolation valves can be opened from the control room. The second note allows up to 4 hours, or until the RCS cold legs exceed 375°F, to place into service ECCS pumps declared inoperable due to LTOP considerations. This note was added since the LTOP setpoint of 330°F is very close to the Mode 3 definition of $\geq 350^\circ\text{F}$. As described in the ITS bases, this note provides operator flexibility to restore the inoperable pump to OPERABLE status. These are Ginna TS Category (v.b.11) changes.
- iv. TS 3.3.1.5.d - This was revised and used as a note for LCO 3.5.2. The specification now only allows ~~878A, 878B, 878C,~~ and 878D to have power installed during MODE 3 for the specific purpose of performing pressure isolation valve testing. Isolation valves ~~878A, 878C, 896A, 896B~~ and 856 must now have DC power removed above MODE 3 or both trains of ECCS will be declared inoperable. This change was made since there is no regularly scheduled testing of ~~878A, 878C, 896A, 896B,~~ and 856 above 350°F. This is a Ginna TS Category (v.a) change.

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- v. LCO 3.5.3 was added which requires one train of SI and RHR during MODE 4. This new requirement is being added to address low probability accidents which may occur during this mode of operation. This is a Ginna TS Category (iv.a) change.
- vi. TS 3.3.1.1.b - The current exception for not requiring the accumulators during hydro tests was not added to the new technical specifications. These hydro tests are performed with RCS temperatures below MODE 3 conditions (i.e., < 350°F). Since the new specification only requires the accumulators when RCS pressure is > 1600 psig during MODE 3, this exception is no longer required. This is a Ginna TS Category (vi) change.
- vii. TS 3.3.1.1.b - The bases for TS 3.3 were revised to update the specified water volume contained in the accumulator with respect to the 50% and 82% levels. The required levels specified in TS 3.3.1.1.b have not been changed, only the corresponding water volumes provided in the bases. The new values are consistent with those used in the accident analysis (see COLR, Table 1). This is a Ginna TS Category (v.c) change.

- viii. TS 3.3.1.1.g - Motor operated isolation valves 851A and 851B were added to new SR 3.5.2.1 since these valves must remain open with AC power removed to ensure the availability of Containment Sump B to the RHR system following a LOCA. The addition of these valves is a conservative change. This is a Ginna TS Category (v.a) change.
- ix. TS 3.3.1.1.h - Check valves 877A, 877B, 878F, 878H, and motor operated isolation valves 878A and 878C were added to this requirement since the valves are required to be tested as PIVs by current Ginna Station TS 4.3.3.3. This provides a more complete specification and is a Ginna TS Category (v.a) change. The listing of valves was also relocated to the bases. This is a Ginna TS Category (iii) change.
- x. TS 3.3.1.1.h and 3.3.1.5 - These requirements were revised to require PIVs to be OPERABLE in MODES 1, 2, 3, and 4 and not just above 350°F (i.e., in MODE 3 and above). Therefore, the plant must now enter MODE 5 within 36 hours if the Required Actions cannot be accomplished. This is a conservative revision which expands the LCO Applicability. This is a Ginna TS Category (iv.a) change.

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xi. TS 3.3.1.5.e - The current requirement allows 12 hours to repair a leaking check valve if the in-series motor operated isolation valve is closed. This was revised to specify that a leaking PIV (check valve or motor operated) must be isolated within 4 hours with a leak tested valve, and that a second leak tested valve must be closed within 72 hours. This is generally a conservative change since a time limit is now specified for isolating the leaking valve and the second isolation valve must now be leak tested. The only exception is that 72 hours is now provided to perform repairs versus 12 hours. The existing allowed repair time is insufficient to perform most leakage repairs and would most likely require a reactor shutdown. Since there are three isolation valves for several flow paths, and the LCO applicability has been expanded to include MODE 4, this change is considered acceptable. This is a Ginna TS Category (v.a) change.

xii. TS 3.3.1.7 and 3.3.1.8 - The exception for allowing the SI pumps to be OPERABLE during DG load and safeguard sequence testing was not added since the new bases allow the pumps to be OPERABLE if a discharge isolation valve is locked closed. Therefore, this exception is not required. Also, these requirements were relocated to the LTOP specification to consolidate all related requirements. These are Ginna TS Category (v.c) and (i) changes, respectively.

- xiii. TS 3.3.1.7.1 and 3.3.1.8.1 - These specifications were converted into Surveillance Requirements consistent with the ITS format and relocated to the LTOP specification to consolidate all related requirements. This is a Ginna TS Category (i) change.
- xiv. TS 3.3.1.8.2 - This requirement was not added since the new bases list the criteria for ensuring that a SI pump is incapable of injecting into the RCS. Limiting the operation to one SI pump when the PORVs provide the RCS vent path is not necessary if the isolation device requires two separate actions before providing an injection path to the RCS. Therefore, operating multiple SI pumps will not pose any threat to overpressurizing the RCS with this isolation. This is a Ginna TS Category (v.c) change.

xv. TS 3.3.2.2 - This was revised to allow both post-accident charcoal filter trains (including the CRFC units which supply them) to be inoperable for up to 72 hours if both containment spray (CS) trains are OPERABLE. This change provides consistency with the accident analyses which demonstrate that either two CS trains, one CS train and one post-accident charcoal filter train, or two post-accident charcoal filter trains are adequate to remove radioactive iodine from the containment atmosphere following a DBA (i.e., each CS train and post-accident charcoal filter train provides 50% of the required iodine removal requirements). However, two CS trains cannot be inoperable since at least one train must operate for containment pressure and temperature control. In addition, two CRFC units can now be removed from service for up to 7 days since the accident analyses only credit two of the four cooling units as being OPERABLE with respect to containment pressure and temperature control. Finally, with one or two CRFC units inoperable and not restored within 7-days, the plant has only 36 hours to reach MODE 5 versus 84 hours due to the importance of maintaining containment pressure and temperature control. These are Ginna TS Category (v.b.12) changes.

xvi. TS 3.3.3.1 - This was revised to ~~only require one of the two CCW heat exchangers to be OPERABLE and to specify that the CCW loop header must also be OPERABLE.~~ The loop header is defined as the section of piping from the discharge of the heat exchangers to the first isolation valve of each supplied component. The loop header then continues from the last isolation valve on the discharge of the supplied component to the suction of the pumps. This is a Ginna TS Category (v.a) change.

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IS 3.3.3.1 - This was revised to allow one CCW heat exchanger to be removed from service for up to 31 days. As discussed in Section C, item 82.1 above, the CCW heat exchangers are 100% redundant and are separated from the CCW pump trains by a section of common piping. The CCW heat exchangers are passive devices such that any failure of a since there is no signal active failure which could fail the redundant heat exchanger is bounded by a failure of the CCW piping in the loop header, 31 days is considered acceptable. The loop header is defined as the section of piping from the discharge of the pumps to the first isolation valve of each supplied component. The loop header then continues from the last isolation valve on the discharge of the supplied component to the suction of the pumps. Since there is no single active failure which must be considered for the heat exchangers, they are considered part of the CCW loop header and only one heat exchanger must be OPERABLE. Requiring the CCW loop header to be OPERABLE provides a clear and concise LCO requirement for operators. These are Ginna TS Category (v.b.13) and (v.a) changes.

Also, there is only one loop header such that a passive failure of the loop header, or the remaining OPERABLE heat exchanger, has the same consequences. This is a Ginna TS Category (v.b.13) change.

xvii. TS 3.3.3.2 - This was revised to allow 72 hours (versus 24 hours) to restore an inoperable CCW pump before requiring a plant shutdown. However, the plant is no longer allowed to remain at Hot Shutdown for 48 hours before requiring additional cooldown to Cold Shutdown conditions. As such, the total time in which a CCW pump can remain inoperable remains the same (i.e., 72 hours) but the plant is not required to begin cooldown activities after 24 hours. The only safety related functions supported by the CCW System are with respect to the RHR, SI, and CS Systems, which all allow 72 hours to restore an inoperable train. Therefore, this change provides consistency within the new specifications. This is a Ginna TS Category (v.c) change.

xviii. TS 3.3.4.1 - This was revised to require that the six sets of motor operated isolation valves used in the SW System to be OPERABLE for the SW System to be considered OPERABLE. Credit is taken for these valves to isolate the nonessential and nonsafety related components within the SW System following a coincident safety injection and undervoltage signal. This is a conservative change which provides a clarification to licensed personnel. This is a Ginna TS Category (v.a) change.

xix.

TS 3.3.4.2 - This was revised to allow one SW train comprised of two pumps and six motor operated valves supplied by the same electrical train to be inoperable for 72 hours before requiring a plant shutdown. Since the SW trains are 100% redundant, removing one of two trains only affects redundancy and does not place the plant outside the accident analyses. Since most other safety functions allow 72 hours for one train to be inoperable (e.g., ECCS trains), this change provides consistency within the new specifications. In addition, this specification was revised to address the scenario if all SW pumps or the SW loop header are inoperable. In this condition, immediate action must be initiated to restore one SW pump or the loop header to OPERABLE status; however, it ~~is~~ may not be prudent to exit the MODE of Applicability since the SW System is required in MODE 5 for decay heat removal. Instead, Required Actions have been provided to require a cooldown to MODE 4 unless the CCW system is incapable of supporting RHR. In this lower MODE 4, AFW is providing for decay heat removal. If AFW were lost, additional time is required before RHR (and consequently SW) would be required. This change is also consistent with the Required Actions for loss of CCW. These are Ginna IS Category (v. - These are Ginna IS Category (v.b.e) changes.

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- xx. TS 3.3.5.1 - This was revised to require the control room emergency air treatment system (CREATS) to be OPERABLE in MODES 1 through 6 and during movement of irradiated fuel assemblies instead of only when RCS is $\geq 350^{\circ}\text{F}$. Current Ginna Station TS 3.5.6 requires that the control room HVAC detection system (i.e., chlorine, ammonia, and radioactivity monitors) be OPERABLE at all times. However, the filtration system is only required to be OPERABLE above 350°F . The filtration system is designed to ensure that dose rates to operators are within the guidelines of GDC 19 in the event of an accident. While dose rates to operators is expected to be lower when the RCS is $< 350^{\circ}\text{F}$, no current analyses exist under these conditions. In addition, failures of the waste gas decay tanks can still occur below 350°F which also require control room isolation. Therefore, the MODE of Applicability was revised to provide consistency within the specifications and the accident analyses. This is a Ginna TS Category (iv.a) change.

- xxi. TS 3.3.5.2 - This was revised to provide requirements for an inoperable filtration train and inoperable dampers. The CREATS dampers isolate the control room in the event of a radiological event while the filtration train filters the control room atmosphere following isolation. The new specification continues to allow the filtration train to be inoperable for 48 hours before requiring a shutdown or placing the control room in the emergency radiation mode (i.e., CREATS Mode 6). If one of the two redundant dampers in each outside air flow path is inoperable, the new specifications allow 7 days to restore the damper to OPERABLE status similar to restoring one train of redundant CREFS in NUREG-1431. If both dampers are inoperable, the plant must enter LCO 3.0.3 since the control room can no longer be isolated. If both dampers are lost in MODES 5 or 6, or during fuel movement, then fuel movement and CORE ALTERATIONS must be suspended immediately. These changes provide consistency with the accident analyses and NUREG-1431. These are Ginna TS Category (v.a) changes.

14. Technical Specification 3.4

- i. TS 3.4.1 - This was revised to specifically require that all MSSVs be tested prior to entering MODE 2 versus the current wording which allows the MSSVs to be removed for testing at any time. This change is consistent with current operating practices and ensures that the MSSVs are OPERABLE before the reactor goes critical but allows the MSSVs to be tested under hot conditions (i.e., $\geq 350^{\circ}\text{F}$). In addition, the MSSV setpoints were added to the new specification since these are assumptions within the accident analyses. These are Ginna TS Category (v.a) changes.

- ii. TS 3.4.2.1.b - This was revised to be consistent with the accident analysis assumptions as discussed in the new bases. Essentially, the accident analyses treat the preferred AFW System as four trains (i.e., two motor driven trains and two turbine driven trains) such that each SG receives flow from two AFW trains. Therefore, the failure of both motor driven trains or the turbine driven train (or both flowpaths) has the same consequence (i.e., loss of one train to each SG). Since the turbine driven train is allowed to be inoperable for up to 72 hours per TS 3.4.2.2.a (and NUREG-1431), this specification was revised to allow both motor driven AFW pumps to be inoperable for up to 72 hours. In addition, if both AFW trains to a common SG are inoperable, the new specifications allow 4 hours to restore at least one train before requiring a controlled cooldown. A time limit for being in this configuration is necessary since no AFW would be available in the event of a HELB which affects the only SG able to receive AFW. Requiring an immediate cooldown in this configuration is not considered prudent since AFW provides for decay heat removal in lower MODES. These are Ginna TS Category (v.b.14) and (v.a) changes, respectively.
- iii. TS 3.4.2.3 - This was revised to require that the SAFW cross-tie be available when the SAFW System is required to be OPERABLE. This change is required since the accident analyses credit the use of the cross-tie for HELBs with a failure of one SAFW pump. Each cross-tie motor operated valve is considered part of the SAFW train which shares the same electrical power source. This is a Ginna Station TS Category (v.a) change.
- iv. TS 3.4.3 - The requirement for SW suction for the AFW and SAFW pumps were relocated to the LCO for these pumps. The CSTs provide the preferred source of condensate to the preferred AFW pumps while the SW System is the safety related source for both the preferred and standby AFW systems. The relocation of the need for a SW supply to the AFW pumps within technical specifications does not reduce the requirement. Instead, the change provides consistency within the new specifications and is easier for licensed personnel to understand. This is a Ginna TS Category (i) change.

- v. TS 3.4.3 - This was revised to require that a backup source of condensate be verified within 4 hours when the CSTs are inoperable versus demonstrating the operability of the SW System. Specifying a time limit for verifying the backup condensate source is a conservative change which now provides a clear and concise requirement for plant operators. Revising the Actions to allow any alternate source to be used as a backup source provides additional operational flexibility since other condensate sources than the SW System can be used if necessary. These sources are described in the bases for new LCO 3.7.6. These changes are consistent with NUREG-1431 and are Ginna TS Category (v.a) changes.

15. Technical Specification 3.5

- i. The following changes were made to TS 3.5.1 or Table 3.5-1:
 - a. Table 3.5-1, Columns 1, 2, and 3 - The columns for the "Total Number of Channels," the "Number of Channels to Trip," and the "Minimum Operable Channels" were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the OPERABILITY of the instrumentation and were relocated to the bases or are adequately described in the UFSAR. This is a Ginna TS Category (iii) change.
 - b. Table 3.5-1, Column 6 - The column for the "channel operable above" was revised consistent with the changes to the Mode table definitions in ITS Chapter 1.0. Changes to the Applicability different from those discussed in Chapter 1.0 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (vi) change.
 - c. Table 3.5-1, Functional Unit #15 - The trip Function was not added to the new specifications. Removal of this trip function is justified in Reference 44 which shows that based on the offsite power system configuration, this trip Function is not applicable to Ginna Station. Therefore, this trip Function was relocated to the TRM. This is a Ginna TS Category (iii) change.

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d. Table 3.5-1, Action Statement #1 for Functional Unit #1 - This action was revised to add requirements for operability of the Manual Reactor Trip function in Modes 3, 4, and 5 when the reactor trip breakers rods are closed not fully inserted and the rod control system is capable of rod withdrawal (LCO 3.3.1, Condition C). These actions ensures the plant is placed in a condition in which the trip function is no longer required for the associated modes of operation. This is a Ginna TS Category (vi) change.

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e. Table 3.5-1, Note 1 for Functional Units #2, #3, and #4 - The notes or remarks which describe an operational detail that are not directly related Unit #11 - This was revised to the OPERABILITY of the instrumentation were not added add the requirements for Turbine Trip on Turbine Stop Valve Closure since this was not in the CTS. These details were relocated to the bases or The required actions with inoperable channels are adequately described in the UFSAR the same as that for the Turbine Trip on Low Autostop Oil Pressure. This is a Ginna TS Category (iii) change (iv).

f) change.

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Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 72 hours (rather than 1 hour).

Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This change is discussed and justified action was revised to allow an inoperable channel to be placed in Reference 30 the tripped condition within 6 hours (rather than 1 hour). This change is a Ginna TS Category (v) discussed and justified in Reference 30. This is a Ginna TS Category (v.15) change.

g) change.

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Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This action was revised to allow an inoperable channel to be bypassed for up to 12 hours (rather than 2 hours) during surveillance testing.

Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This change is discussed and justified in Reference 30 action was revised to allow an inoperable channel to be bypassed for up to 4 hours (rather than 2 hours) during surveillance testing. This change is a Ginna TS Category (v) discussed and justified in Reference 30. This is

a Ginna TS Category (v.15) change.

h15) change.

~~Table 3.5-1, Column 4 - This requirement was revised to associate the permissive (or bypass) details with the specific permissive (or interlock) numbers and to clarify the applicability of the Function with an associated Mode.~~

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Table 3.5-1, Column 4 - This requirement was revised to associate the permissive (or bypass) details with the specific permissive (or interlock) numbers and to clarify the applicability of the Function with an associated Mode (see ITS Table 3.3.1-1, FU #16). The details of the permissible bypass conditions for the associated Functions are discussed in the UFSAR and ITS Bases. Changes to the Applicability of a Functional Unit different from those discussed in Column 4 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (v.c) change.

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i. ~~Table 3.5-1, Action Statement #2 (Channel Operable above column for Functional Unit #2 ("high setting"))~~ This action was revised to add a requirement to either reduce Thermal Power to less than or equal to 75% RTP within 12 hours or to perform a flux map every 24 hours (consistent with SR 3.2.1.2 Units 7, 10, 14 and SR 3.2.2.2) 15. These requirements are in addition ~~this was revised to the requirement to place the channel in the tripped condition within 72 hours as discussed in Section D, item 15~~ change the MODE of Applicability to 8.5% RTP versus 5% RTP. ~~The permissive which enables these functions to be OPERABLE is set at 8.5% RTP per CTS 2.3.2.1.f. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. Performing a flux map compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued operation at power levels above 75% RTP. This is a Ginna TS Category (iv.a) change.~~

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j. ~~Table 3.5-1, Action Statement #3 for Functional Unit #3~~ This action was revised to clarify the applicability of the intermediate range neutron flux to correspond to the specific permissives with either one or two channels inoperable. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. Therefore, this change provides consistency within the CTS. Performing this is a flux map compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued operation at power levels above 75% RTP. This is a Ginna TS Category (v.a) change.

a) change.

~~Table 3.5-1, Action Statement #3 and #4 for Functional Units #2, #3, and #4 - These actions were revised to clarify the applicability of the intermediate range neutron flux and source range neutron flux to correspond to the specific permissives. The NIS intermediate range neutron flux channels must be OPERABLE when the power level is above the capability of the source range and below the capability of the power range. The associated Required Actions ensure the plant is no longer in the applicable condition through controlled power adjustments and taking into account the low probability of an event during the period that may require the protection of the NIS trip. This change supersedes that proposed in Reference 61. This is a Ginna TS Category (v.a) change.~~

k. Table 3.5-1, Action Statement #4 for Functional Unit #4 - This action was revised to clarify the Applicability and add associated Required Actions for inoperable SRMs. For Mode 2 below the permissive and

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only one SRM OPERABLE, the plant would not be required to shut down positive reactivity additions must stop immediately and the inoperable channel restored in 48 hours consistent with current TS. However, with two SRMs inoperable the plant would be required to immediately open the RTBs. For Modes 3, 4, and 5, with the RTBs open CRD incapable of rod withdrawal or all rods not fully inserted, an additional action (LCO 3.3.1, RA L was added that requires the performance of a SDM verification. 2) was added that requires the performance of a SDM verification. These clarifications and additional restriction ensure the plant is no longer in the applicable condition or is in a more stable condition. This is a Ginna TS Category (iv.a) change.

1. Table 3.5-1, Action Statement #5 for Functional Units #8, #9, #10 ("low flow in one loop"), #11 and #13 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 72⁶ hours (rather than 1 hour). ~~This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~

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~~m. Table 3.5-1, Action Statement #5 for Functional Units #8, #9, #10 ("low flow in one loop"), #11 and #13 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 12 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two out of three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~

~~nn. Table 3.5-1, Action Statement #6⁵ for Functional Units #8, #9, #10 ("low flow in both loops")^{one loop}, #14^{#11} and #15^{#13} - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel to be placed in the tripped condition within 72^{for up to 4} hours (rather than 1 hour) in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~

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~~oo. Table 3.5-1, Action Statement #6 for Functional Units #10 ("low flow in both loops"), and #14 and #15 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 12^{to be} placed in the tripped condition within 6 hours in order to perform surveillance testing of other channels (rather than 1 hour). This change is discussed and justified in Reference 30. This is a~~

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GINNA TS Category (v.b.15) change

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Table 3.5-1, Action Statement #6 for Functional Units #10 ("low flow in both loops"), and #14 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

- p. Table 3.5-1, Functional Unit #16 - This was revised to relocate the QPTR Monitor OPERABILITY requirements to Chapter 3.2. In addition, requirements were added to verify with a calculation that the QPTR is within limits every 24 hours when the Quadrant Power Tilt Monitor is inoperable and THERMAL POWER is $< 75\%$ RTP and to verify with a full core flux map that the core power distribution is acceptable every 24 hours when the Quadrant Power Tilt Monitor is inoperable and THERMAL POWER is $\geq 75\%$ RTP. These are Ginna TS Category (i) and (iv.a) changes, respectively.
- q. Table 3.5-1, Functional Unit #17 - The trip function requirement for the Circulation Water Flood Protection was not added. The Circulation Water Flood Protection instruments only provide an anticipatory turbine trip and is not assumed in the Ginna Station safety analysis. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- r. Table 3.5-1, Functional Units #18 and #19 - The Functional Unit applicability was revised to require the instruments to be applicable in all modes associated with DG operability. This ensures that the DG can perform its function on a loss of voltage or degraded voltage to the 480 V buses. This is a Ginna TS Category (iv.a) change.
- s. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This Completion Time is sufficient to allow restoration of the channel and takes into account the redundancy of the trip channels, and the low probability of an event requiring a LOP start occurring during this interval. This is a Ginna TS Category (v.b.16) change.



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t. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel (consistent with LCO 3.0.5) in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with the associated logic. Bypassing the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. Additionally, a note was added clarifying that entry into the associate Conditions and Required Actions can be delayed for up to 4 hours for performance of required surveillance. Entering DG actions during testing is not necessary since the Completion Times for an inoperable DG is much greater than the time to perform the SR (72 hours vs 6 hours). The SR Note time of 6 hours takes into account the redundancy of the trip channels and the low probability of an event requiring a LOP start occurring during this interval. This is a Ginna TS Category (v.b.17) change.

u. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to replace the current shutdown actions with a requirement to restore channels to an OPERABLE status or to enter the applicable conditions for an inoperable DG. The actions of new LCO 3.8.1 and LCO 3.8.2 provide for adequate compensatory actions to assure plant safety. The loss of the minimum required loss of voltage or degraded voltage channels (one bus) should result in actions that are no more restrictive than actions for the loss of one DG. This is a Ginna TS Category (iv.b.1) change.

v. Table 3.5-1, Functional Unit #18 and #19 - The number of channels was reformatted to require only two undervoltage channels per bus versus two channels of the loss of voltage function and two degraded voltage function per bus. The bus undervoltage design is a one-out-of-two taken twice logic such that one degraded voltage channel and one loss of voltage channel comprise each of the two undervoltage channels. However, due to the system design, if either of the degraded voltage or loss of voltage functions is inoperable, the entire undervoltage channel must be tripped (i.e., both the degraded voltage and loss of voltage functions are tripped). This change provides greater clarity to the operators without any reduction in the system requirements. This is a Ginna TS Category (v.b.18) change.

- w. LCO 3.3.1, Table 3.3.1-1, Function #10 was added for the RCP Breaker Position. This function anticipates the Reactor Coolant Flow - Low trips by monitoring each RCP breaker position to avoid RCS heatup that would occur before the low flow trip actuates. The function ensures that protection is provided against violating the DNBR limit due to loss of flow in either a single loop or two loop configuration. This is a Ginna TS Category (iv.a) change.
- x. LCO 3.3.1, Table 3.3.1-1, Function #14 was added for the SI Input from ESFAS. This function ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. A reactor trip is initiated every time an SI signal is present. This is a Ginna TS Category (v.a) change.
- y. Table 3.5-1, Functional Unit #20 and associated Action Statement #14 - This requirement was reformatted to separately denote the Reactor Trip Breakers, the Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, and the Automatic Trip Logic functions (LCO 3.3.1, Table 3.3.1-1, Functions #15, #16, and #17). This is a Ginna TS Category (vi) change.
- z. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Automatic Trip Logic) - This action was revised to allow 6 hours to restore the channel to OPERABLE status in Modes 1 and 2 prior to initiating a plant shut down to Mode 3 (new LCO 3.3.1, Condition Q)(169). The restoration time of 6 hours is reasonable considering that the remaining OPERABLE channel is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.18) change.
- aa. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker) - This action was revised to allow 1 hour to restore the RTB to OPERABLE status in Modes 1 and 2 prior to initiating a plant shut down to Mode 3 (new LCO 3.3.1, Condition R)(169). The restoration time of 1 hour is reasonable considering that the remaining OPERABLE RTB is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.19) change.

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bb. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Automatic Trip Logic) - This action was revised to allow 48 hours to restore the channel to OPERABLE status in Modes 3, 4, and 5 prior to initiating action to open the RTBs (new LCO 3.3.1, Condition C-1). The restoration time of 48 hours is reasonable considering that the remaining OPERABLE channel is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.20) change.

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cc. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker) - This action was revised to allow 48 hours to restore the breaker to OPERABLE status in Modes 3, 4, and 5 prior to initiating action to open the RTBs (new LCO 3.3.1, Condition C-1). The restoration time of 48 hours is reasonable considering that the remaining OPERABLE breaker is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.20) change.

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dd. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms) - This action was revised to only allow 1 hour to open the RTBs following the action to restore the RTB to OPERABLE status in Modes 3, 4, and 5 (new LCO 3.3.1, Condition C-1). The current Ginna Station TS allows 6 hours to perform this action but takes into account a shut down from Modes 1 and 2. The 1 hour provides sufficient amount of time to accomplish the action in Modes 3, 4, and 5 in an orderly manner. This is a Ginna TS Category (v.a) change.

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ee. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms) - This action was revised to specify a limit of 2 hours to bypass the RTB for surveillance testing and 86 hours to bypass the RTB for maintenance on undervoltage or shunt trip mechanisms (new LCO 3.3.1, Condition R, Notes 1 and 2). The current Ginna Station TS for bypassing during maintenance does not specify a time limit. The ITS would set a limit on this time. This is a Ginna TS Category (iv.a) change.

ii. The following changes were made to TS 3.5.2, Table 3.5-2, or Table 3.5-4:

- a. TS 3.5.2.2, 3.5.2.3 and Table 3.5-2, Columns 1, 2, and 3 - The details describing the operability acceptance criteria for Trip Setpoints including the columns for the "Total Number of Channels," the "Number of Channels to Trip," and the "Minimum Operable Channels" were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the operability of the instrumentation and were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.
- b. Table 3.5-2, Column 6 - The column for the "Channel Operable Above" was revised consistent with the changes to the Mode table definitions in ITS Chapter 1.0. Changes to the Applicability different from those discussed in Chapter 1.0 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (vi) change.
- c. Not used Table 3.5-2, Functional Unit #1.b - The Mode of Applicability was revised to be RCS > 200°F. The SI High Containment Pressure Function is used to actuate containment isolation below 350°F such that this function must be operable. The Manual SI Function does not actuate Containment Isolation while the remaining functions are blocked when RCS pressure is < 2000 psig. This is a Ginna TS Category (v.a) change.
- d. Table 3.5-2, Functional Units #1.c and #1.d - The notes or remarks which describe operational details for the Pressurizer Pressure interlock, were reformatted as Mode Applicabilities and default conditions in the new specifications. A new SR 3.3.2.6, was added to specifically denote the operability requirements for the Pressurizer Pressure interlock. This is a Ginna TS Category (iii) change.

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- e. Table 3.5-2, Action Statement #9 for Functional Units #1.b, #1.c, #1.d, #3.b.i, #5.c and #6.b - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to ~~12~~ hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

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- f. Table 3.5-2, Action Statement #9 for Functional Units #1.b, #1.c, #1.d, #3.b.i, #5.c and #6.b - This action was revised to allow an inoperable channel to be placed in the tripped condition within 72⁶ hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- g. LCO 3.3.2, Functional Units #1.b, #2.b, #3.b, #4.b, #5.a, and #6.a, "Automatic Actuation Logic and Actuation Relays," were added for the ESFAS Instrumentation. Actuation logic consists of all circuitry housed within the actuation subsystems, including relay contacts responsible for actuating the ESF equipment. This is merely a presentation change to the Technical Specifications as this logic circuitry is assumed within the operability of the specific Functions. Additionally, the automatic actuation logic and actuation relays for various Functions are required OPERABLE in Mode 4 to support system level manual initiation. This is a Ginna TS Category (iv.a) change.
- h. Table 3.5-2, Action Statement #12 for Functional Unit #3.c - The action associated with this Function was revised to allow an inoperable channel to be placed in the tripped condition within 48 hours (rather than 1 hour). ~~This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~
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~~i. Table 3.5-2, Action Statement #11 for Functional Unit #2.b - The action associated with this Function was revised to replace the limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 12 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two out of three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~
- ~~j. Table 3.5-2, Action Statement #11 for Functional Unit #2.b - The action associated with this Function was revised to allow an inoperable channel to be placed in the tripped condition within 72 hours (rather than 2 hours). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.~~

ki. Table 3.5-2, Action Statement #11 for Functional Unit #3.2.ab - The requirements action associated with this Function was revised to replace the limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for the Auxiliary Feedwater Manual Initiation were not added up to 4 hours in order to perform surveillance testing of other channels. The individual AFW pump instrument requirements only provide a manual function which is not assumed in current requirement limits the Ginna Station safety analysis ability to perform channel functional tests on OPERABLE channels for Functional Units with two out-of-three logic. These instruments do not monitor parameters which are initial assumptions for providing a note to bypass the inoperable channel provides a DBA or transient, do not identify sufficient timeframe to perform the required surveillance testing in a significant abnormal degradation of the reactor coolant pressure boundary, safe and do not provide any mitigation of a design basis event orderly manner. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and this change is relocated to the TRM discussed and justified in Reference 30. This is a Ginna TS Category (iii) change (v.b.15) change.

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ji. Table 3.5-2, Action Statement #11 for Functional Unit #2.b - The action associated with this Function was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 2 hours). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

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ki. Table 3.5-2, Functional Unit #3.a and #3.f - The actions for an inoperable Manual Initiation channel for the AFW and SAFW Systems was revised from restoring operability in 48 hours to declaring the associated pump inoperable. The Manual Initiation channels for these functions actually consist of switches in the control which only actuate one pump train. There is no switch for complete actuation of all AFW or SAFW pumps. Therefore, entering the pump inoperability requirements is consistent with the actions if the AFW pump were declared inoperable. This is a Ginna TS Category (v.b.51) change.

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1. Table 3.5-2, Action Statement #12 for Functional Units #3.b.ii, #3.c, #5.a, and 5.b - The action associated with these Functions was revised to replace the limitation of operation (tied to the next channel functional test of an OPERABLE channel) to

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allow the bypassing of an inoperable channel for up to 124 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

m. Table 3.5-2, Action Statement #12 for Functional Units #3.b.ii, #5.a, and 5.b - The action associated with these Functions was revised to allow an inoperable channel to be placed in the tripped condition within 72~~6~~ hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

n. Table 3.5-2, Action Statement #6 for Functional Unit #3.e - The action associated with this Function was revised to a more restrictive restoration time of 48 hours for an inoperable channel rather than placing the channel in the tripped condition within one hour. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 48~~30~~. This is a Ginna TS Category (iv.a) change.

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o. ~~Table 3.5-2, Functional Unit #3 Not used.~~

~~f. The requirements for the Standby Auxiliary Feedwater Manual Initiation were not added. Table 3.5-2, Functional Unit #4.2. The individual Standby AFW pump instrument requirements only provide a manual function to the Standby AFW pumps which backup the AFW pumps for the Containment Ventilation Isolation (CVI) Manual Initiation Function were not added. The Ginna Station safety analysis does not model the individual current TS are misleading in that there is no manual CVI initiation function for these pumps. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of instead, CVI is manually initiated by the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event Manual CS function. Therefore, the requirement specified for the removal of this function does not satisfy requirement provides consistency within the NRC Final Policy Statement technical specification screening criteria TS and is relocated greater clarify to the TRM operators. This is a Ginna TS Category (iii) change (v).~~

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~~pc) change. Table 3.5-2, Functional Unit #4.2 and Table 3.5-4, Functional Unit #3.b. The requirements for the Containment Ventilation Isolation Function were not added. The containment ventilation components include the shutdown purge and mini-purge lines. These lines are automatically isolated on a containment isolation signal from SI. The R-29 and R-30 instruments are not assumed in the Ginna safety analysis as ESFAS isolation functions. These instruments are, however, required to perform a post accident monitoring function in accordance with Regulatory Guide 1.97 and are retained in new LCO 3.3.3. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event. Therefore, the Manual Isolation and High Containment Radioactive Functions do not satisfy the NRC Final~~

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~~Policy Statement technical specification screening criteria and are relocated to the TRM. The Manual Spray and Safety Injection Functions are deleted since these functions are duplicated by other Functional Units. This is a Ginna TS Category (iii) and (ii) change, respectively.~~

- q. Table 3.5-4, Functional Units #1.b, #1.d, and #2.b - These Functional Unit Allowable Values were revised to reflect the actual values used in the accident analyses. This is a Ginna TS Category (v.c) change.

r. Table 3.5-4, Functional Units #7.a and #7.b - The Trip Setpoint for the loss of voltage and degraded voltage functions were revised to provide a minimum value. Criteria for the establishment of equivalent values based on measured voltage versus relay operating time was relocated to the bases for new LCO 3.3.4). This is a Ginna TS Category (iii) change.

s. Table 3.5-4, Notes 1 and 2 for Functional Units #6.a and #6.c - The notes which describe design details for the Steam Generator Water Level - Low Low Function and Loss of 4 kV Function were not added. These details are relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.

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Table 3.5-2, Functional Unit #4.2b and Action 8 to Table 3.5-5 - The actions for an inoperable CVI radiation monitor were revised to allow 4 hours to isolate the affected penetration. Current TS Table 3.5-2 does not provide a time limit, only that the valves are to be closed while Table 3.5-5 provides 1 hour to perform this action. The time limit is consistent with NUREG-1431 and the Completion Times for an inoperable containment isolation valve which the CVI signal actuates. This is a Ginna TS Category (v.b) change.

- iii. The following changes were made to TS 3.5.3 or Table 3.5-3:
- a. TS 3.5.3.2, TS 3.5.3.3, and Table 3.5-3, Columns 1 and 2 - The columns for the "Total Required Number of Channels," and the "Minimum Channels Operable," were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the operability of the instrumentation and were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.
 - b. TS 3.5.3.2 - The restoration time requirement of 7 days for one inoperable channel (for Functions with two channels) was revised to 30 days. The 30 day Completion Time was revised based on industry operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument, and the low probability of an event requiring PAM instrumentation during this interval. This is a Ginna TS Category (v.b.21) change.
 - c. TS 3.5.3.2 - The action for one channel inoperable for more than 7 days (for Functions with two channels) was revised from requiring a plant shutdown to requiring a Special Report. Due to the passive function of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out-of-service instrumentation. This is a Ginna TS Category (v.b.21) change.

- d. TS 3.5.3.3 - The restoration time requirement of 48 hours for two inoperable channels was revised to 7 days. The 7 day Completion Time was revised based on industry operating experience and takes into account the availability of alternate means to obtain the required information and the low probability of an event requiring PAM instrumentation during this interval. This is a Ginna TS Category (v.b.21) change.
- e. Table 3.5-3 - The Post Accident Monitoring Instrumentation Functions required by this specification were revised to include only RG 1.97, Type A and Category I variables. These functions are denoted in UFSAR Table 7.5-1 and have been previously reviewed and approved by the NRC (Ref. 59). This is a Ginna TS Category (iv.a) change.
- iv. TS 3.5.4 and Table 3.5-6 - The requirements for radiation accident monitoring instrumentation, provided to monitor radiation levels in selected plant locations following an accident, were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.
- v. TS 3.5.6.1 - The requirements for the chlorine gas and ammonia gas instrumentation monitors for control room habitability were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.
- vi. LCO 3.3.5, Conditions B and C, were added for the Control Room Emergency Air Treatment System (CREATS) actuation instrumentation. These new requirements specify Required Actions for various modes of operation when the CREATS isolation dampers cannot be placed in the emergency radiation protection mode. This is a Ginna TS Category (iv.a) change.

vii. TS 3.5.6.2 - The requirement for one detection system inoperable has been revised to allow more than one channel inoperable with an action to isolate the control room in one hour. Even with a loss of Function of the automatic actuation logic, the CREATS may still be capable of being manually isolated within 1 hour and performing its safety function. This is a Ginna TS Category (v.c) change.

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viii. TS 3.5.5 and Table 3.5-5 - The requirements for radioactive effluent monitoring instrumentation which ensures that the limits of TS 3.9.1.1 and 3.9.2.1 are not exceeded were not added (except for R-11 and R-12 which support Containment Ventilation Isolation). No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable or, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively.

ix. Table 3.5-5 - The Mode of Applicability for R-11 and R-12 operability was revised from "during shutdown purges" to MODES 1, 2, 3, and 4, and during MODE 5 when required by LCO 3.9.3, "Containment Penetrations." This is a conservative change which ensures that the gaseous and particulate radiation monitors are operable in all Modes in which the mini-purge or shutdown purge systems can be used. This is a Ginna TS Category (v.a) change.

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16. Technical Specification 3.6

i. TS 3.6.1 - The text allowing closed containment isolation valves to be opened on an intermittent basis under administrative controls was relocated to a LCO Note consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.

ii. TS 3.6.2 - The Applicability for maintaining containment pressure within limits was revised from reactor criticality to MODE 4. This change is necessary to provide consistency with the requirements for containment integrity (i.e., LCO 3.6.1) since exceeding these pressure limits could result in an overpressure of containment if an accident were to occur. This is a Ginna TS Category (iv.a) change.

iii) Also, the time allowed to restore containment pressure was changed from 24 hours to 8 hours. — TS 3.6.3 —

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The title for this LCO change was revised from "containment isolation boundary" to "barrier" which provides greater consistency with the bases for NUREG-1431. It normally takes a short period of time to restore containment pressure to within limits. In addition, three new requirements were added. This is a Ginna TS Category (v). The first requires that a penetration with both containment barriers inoperable be isolated within 1 hour versus 4 hours. — change.

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TS 3.6.3 - Three new requirements were added. The first requires that a penetration with both containment boundaries inoperable be isolated within 1 hour versus 4 hours. This change provides consistency with TS 3.6.1 since containment integrity is potentially violated. As such, verification of continued acceptable containment leakage must be initiated immediately if both barriers are declared inoperable. In addition, new requirements with respect to an inoperable airlock (including the use of an airlock with an inoperable door or interlock mechanism) and containment mini-purge penetrations with isolation valves that exceed their leakage rate acceptance criteria were added. The new requirement for the airlocks specifies that an inoperable airlock door (including an inoperable interlock mechanism) must be isolated within 1 hour and locked closed within 24 hours. However, a dedicated individual can be used in the case of an inoperable interlock mechanism to allow entry and exit through the airlock. The new specification provides specific Required Actions in the event that current Ginna Station TS 4.4.2.44 4.2.3.c is exceeded. The new requirement for the mini-purge penetrations specifies that the affected penetration must be isolated within 24 hours if an isolation valve exceeds its leakage limit. These new requirements provide added assurance that penetrations which can provide direct access to the outside environment are addressed quickly when their isolation barriers become inoperable. This is a Ginna TS Category (iv.a) change.

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iv.

TS 3.6.3 - The use of a closed system to isolate an inoperable containment isolation barrier for up to 72 hours was added to this specification. Consequently, a closed system which must be OPERABLE to meet this specification can be used to isolate a failed isolation barrier for a limited period of time. Also, isolation devices located outside containment that were used to isolate a failed containment isolation valve are required to be verified closed once every 31 days. For isolation devices inside containment, they must be verified closed upon entry into MODE 4 from MODE 5 if it has not been performed within the last 92 days. These are Ginna TS Category (v.b.22) changes.

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v.

TS 3.6.5 - This was the reasons for opening the mini-purge valves above 200°F were relocated to the bases for ITS 3.6.3 since ~~it does~~ these do not meet any of the four criteria and ~~does~~ not specify any Required Actions. Operation of the Mini-Purge System is performed under procedures such that its use is strictly controlled. Placing this information in the bases also provides similar control under 10 CFR 50.59 (i.e., the Bases Control Program). This is a Ginna TS Category (iii) change.

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- vi. TS 3.6 - A new requirement was added which specifies that the average containment air temperature shall be $\leq 120^{\circ}\text{F}$ above MODE 5. This temperature limit is necessary to ensure that the resulting containment temperature following a DBA is within the assumptions used for environmental qualification of components within containment. If the average containment air temperature is $> 120^{\circ}\text{F}$, it must be restored within 24 hours. This is a Ginna TS Category (iv.a) change.
- vii. TS 3.6 - A new requirement was added which requires the hydrogen recombiners to be OPERABLE in MODES 1 and 2. The hydrogen recombiners are assumed in the accident analyses to be used to prevent a hydrogen explosion within containment that could overpressurize the containment structure. The new LCO allows 30 days to restore an inoperable recombiner and 7 days to restore two inoperable recombiners if the Mini-Purge System is OPERABLE. In addition, the plant can enter MODES 1 and 2 with an inoperable hydrogen recombiner. This is a Ginna TS Category (iv.a) change.
- viii. TS 3.6.4.1 and TS 3.6.4.3 - The Applicability for the hydrogen monitors was revised to include Mode 3 requirements. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in Modes 1, 2, and 3. This is a Ginna TS Category (iv.a) change.
- ix. TS 3.6.4.2 - The action for one channel inoperable for more than 30 days was revised from requiring a plant shutdown to requiring a Special Report. Due to the passive function of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out-of-service instrumentation. This is a Ginna TS Category (v.b.21) change.
- x. 221 TS 3.6.1.b and ~~TS 3.6.1~~ The requirement describing the specific applicability for containment integrity if the boron concentration is less than 2000 ppm was not added. ~~e~~ The requirement describing No screening criteria apply for this requirement since the specific applicability for containment integrity was not added boron concentration limit is only a requirement for fuel handling accidents. No screening criteria apply for this requirement since containment integrity is not assumed in Since the refueling safety analysis requirements of ITS LCO 3.6.1 immediately stop all fuel movement in this condition, there is no for containment integrity. The fuel handling accident inside containment analysis (UFSAR 15.7.3.3) takes no credit Therefore, the requirements specified for isolation of the containment, containment integrity, nor effluent filtration prior to release this function do not satisfy the

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~~NRC Final Policy Statement technical specification screening criteria and have been deleted. The requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (V. Boron concentration changes in MODE 6 and Required Actions to suspend positive reactivity additions is provided in new LCO 3.9.1b. This is a Ginna TS Category (iii)50) change.~~

17. Technical Specification 3.7

- i. TS 3.7.1.1.b, 3.7.1.1.d, and 3.7.1.1.e - The cold shutdown or refueling requirements (MODES 5 and 6) for the 480 V safeguards buses, batteries and DC trains, and 120 VAC instrument buses were revised from requiring only one train to be OPERABLE to require the necessary train(s) to support all other LCO requirements. Consequently, one or both trains of these systems may be required depending on other system requirements (e.g., RHR). In MODES 5 and 6, sufficient electrical power redundancy must be available to mitigate an event coincident with either a loss of offsite power, loss of all onsite standby emergency power, or a worse case single failure. This change ensures that all necessary electrical support systems are OPERABLE to respond to a DBA or a transient. This is a Ginna TS Category (iv.a) change.
- ii. TS 3.7.1.2 - Cold or refueling requirements (MODES 5 and 6) for the DG fuel oil supply and the battery parameters have been added to provide restoration times for specified conditions consistent with the ITS. These times are sufficient to complete restoration of the degraded parameter prior to declaring the component inoperable and is acceptable based on the low probability of an event during this brief period and the fact that the component remains capable of performing most required functions. This is a Ginna TS Category (v.a) change.
- iii. TS 3.7.2.1.b.2, 3.7.2.2.a, and 3.7.2.2.b - The requirements for two offsite sources were not added. The current actions allow the plant to operate indefinitely with one offsite source inoperable. The new ITS format criteria would not specify these requirements in the TS (i.e., require a component for a MODE change but allow the component to remain inoperable indefinitely once the MODE change is complete). Therefore, these requirements are ~~deleted~~ ~~relocated to the TRM~~. The offsite power sources are further discussed in Reference 32. This is a Ginna TS Category ~~(v.(iii) change. b.23) change.~~
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- iv. TS 3.7.2.2.b.1 - The actions for an inoperable DG have been revised: (1) to eliminate the testing of the OPERABLE DG if, within 24 hours, it can be determined that the OPERABLE DG is not inoperable due to common cause failure, and (2) to require verification of the offsite power circuit to the affected AC distribution train. In addition, the OPERABLE DG must only be tested once during the 7 day allowed outage for the inoperable DG. The revised action for the OPERABLE DG eliminates unnecessary testing during a period in which the plant relies on only one DG. These are Ginna TS Category (iv.b.2) and (v.a) changes.

- v. TS 3.7.2.2.c - The Completion Time for the action to re-energize the 480 V safeguards bus has been revised from 1 hour to 8 hours. The time is consistent with the ITS which assumes not only restoration of the bus but also the associated load centers, motor control centers, and distribution panels which comprise the AC electrical train. This is a Ginna TS Category (v.b.24) change.
- vi. TS 3.7.2.2.d - This was revised to address the scenario with both offsite power and one DG were inoperable. In this condition, no loss of safety function exists since the remaining DG is available to provide power to one ESF train. However, the time in this Condition should be limited due to the potential to lose multiple safety functions if the remaining DG were lost. Therefore, a Completion Time of 12 hours is provided. However, if both offsite power and one DG were inoperable to the same AC electrical train, then the time would be restricted to 8 hours as discussed in Section D, item 17.v above. This is a Ginna TS Category (v.a) ~~changed~~ ⁵⁴ change.

18. Technical Specification 3.8

- i. ~~TS 3-8-13.8 - The applicability was revised from "during refueling operations" to "CORE ALTERATIONS and irradiated fuel assembly movement within containment." and 3.8.3 - The requirements to close containment penetrations during fuel handling in the containment were not added. No screening criteria apply for these requirements. This is an equivalent change since these conditions are not assumed in the refueling safety analysis. Operations can only be related to CORE ALTERATIONS and irradiated fuel assembly movement within containment. The fuel handling accident inside containment analysis (UFSAR 15.7.3.3) takes no credit for isolation of the containment nor effluent filtration prior to release from the containment building. This is a Ginna TS category (v. Therefore, closure of containment penetrations during fuel handling inside containment is not required.) change.~~

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~~The closure of the containment penetrations were established to provide additional margin for the fuel handling analysis and to provide protection against the potential consequences of seismic events during refueling. The dose consequences, however, of the fuel handling accident inside containment analysis is estimated at approximately 30% of 10 CFR 100 limits. This was found to be "well within" limits as documented in the NRC Safety Evaluation Report (SER) dated October 7, 1981 (Ref. 49). The requirements specified for these conditions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.~~



ii.

TS 3.8.1.b - The refueling or MODE 6 requirement for the containment radiation monitors which provide monitoring for personnel safety was not added. No screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the containment radiation monitors are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

- iii. TS 3.8.1.c - The requirement describing the specific applicability of the SRMs was revised. The phrase "whenever geometry is being changed" is covered by the new TS definition of MODE 6. The requirement that one SRM be OPERABLE when core geometry "is not being changed" is covered by the Required Action for one inoperable SRM. This would restrict CORE ALTERATION and positive reactivity additions when core geometry is not being changed. Required Actions were also provided when two SRMs become inoperable or when the audible indication is lost. These new actions require verification of boron concentration every 12 hours and ensures the stabilized condition of the reactor core. These are a conservative revisions and Ginna TS Category (v.a) and (iv.a) changes, respectively.
- iv. TS 3.8.1.e - The requirement describing the specific applicability and frequency of the boron concentration sampling was revised. The phrase "immediately before reactor vessel head removal and while loading and unloading fuel from the reactor" is covered by the new TS definition of MODE 6. This would additionally require boron concentration sampling throughout MODE 6. The sampling frequency, however, was also revised to require sampling every 72 hours. These revisions consider the large volume of the refueling canal, RCS, and refueling cavity and are adequate to identify slow changes in boron concentration. Rapid changes in boron concentration, described in UFSAR 15.4.4.2, are detected by the SRM instrumentation required by new TS 3.9.2. This is a conservative revision and a Ginna TS Category (iv.a) change.
- v. TS 3.8.1.f - The requirement for communication with the control room during CORE ALTERATIONS is not added. No screening criteria apply for this requirement since communications is not part of the primary success path assumed in the mitigation of a DBA or transient. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.
- vi. TS 3.8.1.d (footnote *) and TS 3.8.1.g (footnote *) - The requirement that either the preferred or the emergency power source may be inoperable for each residual heat removal loop is not added. This detail is encompassed in the definition of operability described in new TS 1.1 and the electric power requirements contained in Chapter 3.8. This is a Ginna TS Category (i) change.

vii. TS 3.8.1.c - The requirement to provide SRM audible indication in the containment was not added. No screening criteria apply for this requirement since the monitored parameter (audible indication in containment) is not assumed in the refueling safety analysis. The safety analysis assumes audible indication in the control room which is denoted by new LCO 3.9.2. The audible indication is for personnel safety only. Further, the audible indication is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

viii. TS 3.8.1.iii.1 - The isolation devices which are allowed was revised to include "or equivalent." The use of "or equivalent" for isolation of a containment penetration is consistent with NUREG-1431 and the bases for GTS 3.8 which allow the use of a "material which can provide a temporary ventilation barrier, at atmospheric pressure, for the containment penetrations during fuel movement." Therefore, this is a clarification only to the LCO. This is a Ginna TS Category (v.c) change.

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ix. TS 3.8.2 - The requirement to initiate action "to correct the violated conditions" was not added to the new specifications since this is always an option to exit the Condition. That is, the Condition is not exited, even after completion of the Required Actions, unless either the LCO is met, or the MODE of Applicability is exited. This is a Ginna TS Category (v.c) change.

x. TS 3.8.2 - The requirement to cease "operations which may increase the reactivity of the core" was not added to LCO 3.9.3 (Containment Penetrations) since the basis for isolation of containment is with respect to a fuel handling accident. The reactivity of the core is with respect to a boron dilution event which is adequately addressed by other LCOs. This is a Ginna TS Category (v.b.48) change.

19. Technical Specification 3.9

- i. TS 3.9.1.1 - The requirements for radioactive material released in liquid effluents to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20, Appendix B; Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

ii. TS 3.9.1.2 and TS 3.9.2.4 - The requirements for dose or dose commitment to individuals which results from cumulative liquid effluent discharges during normal operation over extended periods and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I, ~~40 CFR 141, and 40 CFR 190 limits~~ were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, radioactive liquid effluent dose projected value is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

iii. TS 3.9.1.3 - The requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, ~~GDC 60 and 10 CFR Part 50, Appendix I, Section II~~ were not added. ~~D, were not added.~~ No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

iv.

TS 3.9.2.1 - The requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

v. TS 3.9.2.2.a, TS 3.9.2.2.c, and TS 3.9.2.4 - The requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

vi. TS 3.9.2.2.b, TS 3.9.2.2.c, and TS 3.9.2.4 - The requirements for dose due to radioiodine, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days released with gaseous effluents were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, these gaseous effluents doses are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- vii. TS 3.9.2.3 - The requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- viii. TS 3.9.2.5 and TS 3.9.2.6 - The specific requirements for which limit concentration of oxygen in a gas decay tank and the quantity of radioactivity contained in each waste gas decay tank were not added. The level of detail is relocated to Explosive Gas and Storage Tank Radioactivity Monitoring Program described in new Specification 5.5.11 and a more generic description is provided. This is a Ginna TS Category (iii) change.
- ix. TS 3.9.2.7 - The requirements for the solid radwaste system which processes wet radioactive waste and operates in accordance with 10 CFR Part 50, Appendix A, for effluent control were not added. The operability of the system is not assumed in the analysis of any DBA or transient. Further, radioactive waste is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

20. Technical Specification 3.10

- i. TS 3.10.1 - This was revised to include specific actions and Completion Times for cases when the shutdown bank insertion limits and the control bank insertion, sequence, and overlap limits are not within the limits specified in the COLR. These actions require verification within 1 hour that the SHUTDOWN MARGIN is within limits and restoring the associated value to within limits within 2 hours or be in MODE 3 within 6 additional hours. These additions were made to ensure that the control banks and the shutdown bank are available as assumed in the safety analyses. This is a Ginna TS Category (iv.a) change.

- ii. TS 3.10.1.1 - This was revised to include a specific action to initiate boration within 15 minutes when the SHUTDOWN MARGIN is not within limits. The addition of this action ensures that SHUTDOWN MARGIN is monitored and quickly restored within limits. This is a Ginna TS Category (iv.a) change.
- iii. TS 3.10.1.1 and Figure 3.10-2 - These were revised to relocate the SHUTDOWN MARGIN requirements and Figure 3.10-2 to the COLR. SHUTDOWN MARGIN can be used in fuel management and as a variable to solve plant specific problems. SHUTDOWN MARGIN impacts a number of analyses (i.e., uncontrolled boron dilution and steamline break) and is sensitive to many core related parameters such as control bank position, core power level, coolant temperature and cycle specific parameters such as fuel burnup, xenon concentration and boron concentration. The inclusion of SHUTDOWN MARGIN in the COLR provides more flexibility in plant operation, in performing the design, and in obtaining good fuel economics particularly for extended cycle operation. With the SHUTDOWN MARGIN included in the COLR, the core design can be finalized after shutdown so that the actual end of cycle burnup is known which is particularly helpful when the actual burnup differs from the projected value. This is a Ginna TS Category (iii) change.
- iv. TS 3.10.1.2 and TS 3.10.1.3 - These were revised to indicate only low power PHYSICS TEST exceptions for the shutdown and control bank insertion limits. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the insertion limits. This is a Ginna TS Category (vi) change.
- v. TS 3.10.1.3 and Figure 3.10-1 - These were revised to relocate the control rod insertion limits and the sequence and overlap limits to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. This is a Ginna TS Category (iii) change.
- vi. TS 3.10.1.5 - This was not added to the new specifications. None of the PHYSICS TESTS currently performed at Ginna Station currently require a relaxation of the SHUTDOWN MARGIN requirements. Therefore none of these SHUTDOWN MARGIN PHYSICS TESTS exceptions or Required Actions are necessary. This is a Ginna TS Category (vi) change.
- vii. TS 3.10.2.2 - This was revised to remove the low power PHYSICS TESTS exception since new LCO 3.2.1 and LCO 3.2.2 which contain the peaking factor requirements are only applicable in MODE 1. This is a Ginna TS Category (v.a) change.

- viii. TS 3.10.2.3 - This was revised to remove the PHYSICS TEST exceptions for the QPTR. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the QPTR limit and the ITS LCO 3.2.4 which contains QPTR is only applicable in MODE 1 with THERMAL POWER \geq 50% RTP. This is a Ginna TS Category (vi) change.
- ix. TS 3.10.2.8, TS 3.10.2.9 and TS 3.10.2.10 - These were revised to remove the PHYSICS TEST exceptions for AFD. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the AFD limits and the ITS LCO 3.2.3 which contains AFD is only applicable in MODE 1 with THERMAL POWER \geq 15% RTP. This is a Ginna TS Category (vi) change.
- x. TS 3.10.3.1.a - This was revised to reduce the minimum T_{avg} for the rod drop test from 540°F to 500°F. The 500°F temperature is conservative since the water will be slightly denser at the lower temperature which has the potential to slow down the dropped rods. This change would enable the plant to complete the rod drop test at an earlier time during plant startup and is consistent with NUREG-1431. This is a Ginna TS Category (v.a) change.

- xi. TS 3.10.4.1 - This was revised to indicate only low power PHYSICS TEST exceptions for control bank alignment. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the alignment limits. This is a Ginna TS Category (vi) change.
- xii. TS 3.10.4.2 and TS 3.10.4.3 - These were revised to remove conditions of rod inoperability due to being immovable. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if they are immovable. Reference to full length rods was also removed since there are no part length rods in the reactor core. This is a Ginna TS Category (v.c) change.
- xiii. TS 3.10.4.3.2 - This was revised to remove the requirement to declare a misaligned rod inoperable when the rod cannot be restored to within the alignment limits in 1 hour. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if they are immovable. This is a Ginna TS Category (v.a) change.
- xiv. TS 3.10.4.3.2.a - This option for restoring a rod to within alignment was removed from the LCO and relocated to the Bases for ITS 3.1.4 which is controlled under the Bases Control Program. This is a Ginna TS Category (iii) change.
- xv. TS 3.10.4.3.2.b.iii and Table 3.10-1 - These were revised to remove Table 3.10-1 from the specifications. The ITS requires evaluations of accident analysis to be performed to determine that the core limits will not be exceeded during a Design Basis Accident. An evaluation of each of the analyses on Table 3.10-1 may not be required to determine that the core limits will not be exceed. This table was relocated to the TRM. This is a Ginna TS Category (iii) change.
- xvi. TS 3.10.4.3.2.b and TS 3.10.4.3.2.c - These were revised to remove the requirement to reduce the high neutron flux trip setpoint to $\leq 85\%$ RTP when the power level is reduced to $\leq 75\%$ RTP. This required action is deleted based on agreements between the NRC and the owners groups and is consistent with WCAP-13029 (Ref. 50) which states that the safety analyses results would not be significantly affected by changes to their initial assumptions as a result of increased peaking factors caused by rod misalignment. Additionally, the peaking factor limit verification within 72 hours and the re-evaluation of the safety analysis within 5 days that are required by this specification provide further assurance that the assumptions made in the safety analysis are preserved. This is a Ginna TS Category (v.e) change.

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- xvii. TS 3.10.4.4 - This was revised to include an action to verify SHUTDOWN MARGIN or initiate boration within 1 hour when more than one rod is out of alignment. The ITS Bases state that 1 hour is a reasonable time based on the time required for potential xenon distribution and the low probability of an accident. This is a Ginna TS Category (v.a) change.
- xviii. TS 3.10.5.1 - This was revised to add an action statement to clarify that if more than one MRPI is inoperable per group for one or more groups or more than one demand position indicator per bank is inoperable for one or more banks then the plant must enter 3.0.3 immediately. This is a Ginna TS Category (v.a) change.
- xix. TS 3.10.5.2.a - This was revised to allow 4 hours (instead of immediately) to verify rod position. The rod position cannot be determined immediately. It takes time to acquire the data and obtain the results. "Immediately" is considered a start time not a completion time. The ITS Bases state that 4 hours provides an acceptable period of time to verify the rod positions while a reduction to $\leq 50\%$ RTP will avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP with 2 or more rods misaligned. This is a Ginna TS Category (v.c) change.

- xx. TS 3.10.2.1 - This was revised to require measurement of the power distribution after each fuel reloading prior to operation of the plant at or above 75% RTP instead of prior to 50% RTP consistent with ITS. This requirement ensures that the design limits are not exceeded when RTP is achieved, since peaking factors are usually decreased as power increases. Requiring this surveillance at 75% versus 50% still provides the necessary margin to ensure that design safety limits are not exceeded and provides the operator with more flexibility during power ascension following a refueling. This is a Ginna TS Category (v.b.25) change.
- xxi. TS 3.10.2.1 - This was revised to delete the requirement to verify QPTR using movable incore detectors or core exit thermocouples with one power range detector inoperable at THERMAL POWER \geq 75% RTP and replaced with a requirement to perform a flux map to verify that hot channel factors are within limits consistent with ITS. The incore detectors are not used to verify QPTR but rather to verify that the core power distribution is acceptable. Ginna Station does not have 8 pairs of symmetric thimble plugs which are necessary to perform a partial flux map and thus would have to complete a full core flux map to verify that the core power distribution is acceptable. This change is consistent with current interpretations at Ginna Station and is preferred by Ginna Station licensed personnel. This is a Ginna TS Category (v.c) change.
- xxii. TS 3.10.2.2 - This was revised to require the hot channel factors be within limit only in MODE 1. The proposed Applicability does not require the F_Q or $F_{\Delta H}$ limits to be met in MODES 2 - 5 or during refueling. As described in the ITS Bases, F_Q and $F_{\Delta H}$ must be within limits during MODE 1; however, such limits are not necessary in MODE 2 because there is insufficient stored energy in the fuel or being transferred to the coolant to require these limits. This is a Ginna TS Category (v.b.26) change.
- xxiii. TS 3.10.2.2 - This was revised to relocate the limits for $F_Q(Z)$ and $F_{\Delta H}$ and the Figure 3.10-3 to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. This is a Ginna TS Category (iii) change.
- xxiv. TS 3.10.2.2 - This was revised to include an administrative Action to reduce the AFD acceptable operational limits specified in the COLR by the percentage that F_Q exceeds the limit. This is necessary since a change in F_Q can adversely impact AFD limits. A Completion Time of 8 hours is allowed to perform this action. This is a Ginna TS Category (iv.a) change.

xxv. TS 3.10.2.2 - This was revised to allow 72 hours (instead of 24 hours) to reduce the Overpower ΔT and the Overtemperature ΔT trip setpoints when F_Q or $F_{\Delta H}$ is not within limits consistent with NUREG-1431. This section was also revised to include a Completion Time of 72 to reduce the Power Range Neutron Flux High trip setpoints. These actions provide further protection against the consequences of severe transients with unanalyzed power distributions. The 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the initial prompt reduction in THERMAL POWER. This is a Ginna TS Category (v.b.27) change.

xxvi. TS 3.10.2.2 - This was revised to add a Required Action to be in MODE 2 within 6 hours if the Required Actions and associated Completion Times for the Condition when F_Q or $F_{\Delta H}$ is not within limits is not met. This action places the plant in a condition outside of the Applicability requirements for the Hot Channel Factor requirements. The Completion Time of 6 hours is sufficient to reach MODE 2 from full power operation in an orderly manner without challenging plant systems. This is a Ginna TS Category (iv.a) change.



xxvii.

TS 3.10.2.3 and 3.10.2.4 - These were revised to specifically define the Applicability requirements for QPTR as MODE 1 with THERMAL POWER > 50% RTP. This Applicability is consistent with the current requirements for Ginna Station since continued operation is allowed for an unlimited period of time when THERMAL POWER is < 50% RTP. The ITS Bases state that below 50% RTP there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. In addition, the LCO limit of 1.12 was removed since the primary limit of 1.02 will be reached initially and actions will already be in progress to address the tilt. THERMAL POWER will continue to be reduced if the tilt ratio continues to increase. This revision is consistent with the changes made in WCAP-12159 (Ref. 51) and current industry practice. These are Ginna TS Category (v.a) changes.

xxviii. TS 3.10.2.3 - This was revised to limit the THERMAL POWER relative to the percentage of quadrant power tilt, (i.e., limit power to 3% below RTP for each 1% by which the QPTR exceeds 1.00) instead of requiring an immediate action to go below 75% RTP. The reduction to 75% RTP essentially employs a 2% RTP reduction for each 1% the QPTR was above 1.00 up until QPTR equalled 1.12 where a reduction to 50% RTP was required. The proposed change would provide flexibility with the initial reduction, but would require at least a 3% RTP reduction for each 1% QPTR exceeded 1.00. Thus, the proposed change while requiring a smaller reduction for small tilts is more conservative for larger tilts which would suggest a more serious problem. This revision is consistent with the changes made in WCAP-12159 (Ref. 51) and current industry practice. The requirement to measure the hot channel factors when the QPTR exceeds 1.02 is changed from within 2 hours to within 24 hours since the THERMAL POWER is appropriately limited within 2 hours. The 24 hour Completion Time takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. This is a Ginna TS Category (v.a) change.

xxix. TS 3.10.2.4 - This was revised to delete the requirement to identify the cause of the tilt or limit power to < 50% RTP. Identification of the cause of the tilt is not always possible and other actions already underway are adequate to assure safe operation of the plant (e.g., surveillances). This change is consistent with WCAP-12159 (Ref 53). This is a Ginna TS Category (v.b.28) change.

xxx. The following Required Actions were added for the Condition when QPTR is not within the limit: These are Ginna TS Category (iv.a) changes.

- a. A requirement to verify by calculation that the QPTR is within limits and limit power accordingly every 12 hours.
- b. A requirement to recalibrate the excore detectors prior to increasing THERMAL POWER above the limit in TS 3.10.2.3. This action is modified by a Note that requires verification that the hot channel factors are within limits prior to recalibration of the excore detectors.

- c. A requirement to verify F_Q and $F_{\Delta H}$ within limits either within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit in TS 3.10.2.3. This action is modified by several Notes. The first Note clarifies that when the QPTR alarm is due to instrumentation alignment this action does not need to be completed. The second note allows this action to be completed only after the excores have been recalibrated. The third note clarifies that the Completion Time applicable first is the one that must be met.
 - d. A requirement to reduce power to < 50% RTP within 4 hours if the initial Required Actions are not met within the associated completion time. This takes the plant out of the Applicability when the actions are not met and provides an additional action before plant shutdown is required.
- xxxii. TS 3.10.2.5 - This was deleted since the 1.12 QPTR limit no longer applies and the Applicability requirement for QPTR has been revised to > 50% RTP. Actions already in progress (i.e., limiting power by 3% below RTP for each 1% QPTR exceeds 1.00) are sufficient to address the tilt. This is a Ginna TS Category (v.c) change.
- xxxii. TS 3.10.2.7 - This was revised to require a measurement of the target flux difference within 31 EFPD after each refueling instead of within 92 EFPD. This requirement is also modified with a note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling. The change to within 31 EFPD is more conservative than within 92 EFPD and is necessary to perform the initial monthly target flux difference update also required by TS 3.10.2.7. This is a Ginna TS Category (v.a) change.
- xxxiii. TS 3.10.2.8 - This was revised to relocate the AFD target band to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The Applicability requirement was also revised to specify MODE 1 with THERMAL POWER > 15% RTP. As described in the ITS Bases, this Applicability is acceptable because of the low amounts of stored or transferred energy in the lower MODES. The AFD at these lower conditions does not affect the consequences of the design basis events. Additionally, the low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. These are Ginna TS Category (iii) and (v.c) changes, respectively.

- xxxiv. TS 3.10.2.9 - This was revised to specify 15 minutes (instead of immediately) to restore AFD to within the target band and then immediately initiate actions to reduce THERMAL POWER to $< 90\%$ RTP if the AFD is not restored within the initial 15 minutes. This is consistent with the intent of the current Ginna Station technical specifications. "Immediately" is considered a start time not a completion time and 15 minutes is considered a sufficient amount of time to restore AFD within limits without allowing the plant to remain in an unanalyzed condition for an extended period of time prior to a reduction in power. This is a Ginna TS Category (v.c) change.
- xxxv. TS 3.10.2.10a - This was revised to relocate the AFD target band and the acceptable operation limits to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The Applicability requirement was also revised to specify MODE 1 with THERMAL POWER $> 15\%$ RTP. As described in the ITS Bases, this Applicability is acceptable because of the low amounts of stored or transferred energy in the lower MODES. The AFD at these lower conditions does not affect the consequences of the design basis events. Additionally, the low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. These are Ginna TS Category (iii) and (v.c) changes respectively.
- xxxvi. TS 3.10.2.12 - This was revised to require a verification that the AFD is within limits and to log the AFD every 15 minutes with THERMAL POWER $\geq 90\%$ RTP and once every hour with THERMAL POWER $< 90\%$ RTP when the AFD monitor alarm is inoperable instead of every hour for the first 24 hours and every half hour thereafter. This modification reflects the importance of staying within the target band at above 90% RTP and is consistent with the Required Action if the AFD is outside the target band. This is a Ginna TS Category (v.c) change.

21. Technical Specification 3.11

- i. TS 3.11.1 - This was revised to require that the Auxiliary Building Ventilation System (ABVS) be OPERABLE when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. The specific components which are required for the ABVS to be considered OPERABLE were relocated to the bases similar with the structure of NUREG-1431 and the ITS Writer's Guide. The bases for LCO 3.7.10 now require that one of the two 100% capacity Auxiliary Building main exhaust fans, exhaust fan C, the SFP Charcoal Absorber System, and all associated ductwork, valves and dampers be OPERABLE. In addition, TS 3.11.1.c was revised to require a negative pressure within the Auxiliary Building operating floor with respect to the outside environment instead of requiring all doors, windows, and other direct openings between the operating floor area and the outside to be closed. This change provides consistency with assumptions of the fuel handling accident as described in the bases. This change also provides a much clearer specification which is easier for licensed personnel to read and understand without any reduction in actual requirements. These are Ginna TS Category (i) and (v.a) changes, respectively.
- ii. TS 3.11.2 - The requirement to continuously monitor radiation levels in the SFP area was not added to the new specifications. No screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the SFP radiation levels only provide a backup source to a SFP problem. Other LCOs provide adequate verification of SFP primary indications (i.e., level and boron concentration) which ensure that all accident analysis assumptions are met. Since a fuel handling accident can only occur as a result of fuel movement, personnel would be stationed within the Auxiliary Building and immediately aware of a problem. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- iii. TS 3.11.3 and 3.11.5 - The heavy load restriction for movement of loads over the SFP was not added to the new specifications. No screening criteria apply for this requirement because the heavy load limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This change is consistent with WCAP-11618 (Ref. 52) and is a Ginna TS Category (iii) change.

- iv. TS 3.11.4 - The SFP water temperature limit was not added to the new specifications. No screening criteria apply for this requirement because the SFP water temperature limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.

22. Technical Specification 3.12

- i. TS 3.12.1 - The requirement for the number of thimbles per quadrant required to OPERABLE during recalibration of the excore axial off-set detection system was not added to the new specifications. The requirements for this surveillance are not an initial assumption of any DBA or transient analysis. Therefore, this specification does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.

23. Technical Specification 3.13

- i. TS 3.13 - The requirements for snubbers operability were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to the Inservice Testing Inspection Program described in new Specification 5.5.8 and more generic program description is provided. This is a Ginna TS Category (iii) change.

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24. Technical Specification 3.14

None.

25. Technical Specification 3.15

- i. TS 3.15.1 - The LTOP exception during secondary side hydrostatic testing was relocated as a NOTE to new LCO 3.4.12. This is a Ginna TS Category (v.c) change.
- ii. TS 3.15.1 - The PORV setpoint during LTOP conditions was relocated to the PTLR consistent with LCO 3.4.12. This is a Ginna TS Category (iii) change.
- iii. TS 3.15.1 - The accumulators are now required to be isolated when the accumulator pressure is greater than the maximum RCS pressure for existing cold leg temperatures as specified in the PTLR consistent with Condition C of LCO 3.4.12. This new requirement prevents an accumulator from overpressurizing the RCS and causing an actuation of the LTOP System. The operator is instructed to isolate or depressurize the affected accumulator under these conditions. This is a Ginna TS Category (iv.a) change.

- iv. TS 3.15.1.1 - A new requirement was added when a PORV is inoperable during MODES 5 and 6 due to the increased consequences from an overpressurization event under these conditions. This new requirement specifies that the PORV must be restored to OPERABLE status within 72 hours. The limit of 72 hours with one PORV inoperable is consistent with the allowed outage time for one train of ECCS equipment during MODES 1, 2 and 3. This is a Ginna TS Category (iv.a) change.
- v. TS 3.15.1.3 - The reporting requirement for the low temperature overpressure protection (LTOP) system operation was revised. The reporting requirement to include documentation of all challenges to the pressurizer power operated relief valves is detailed in proposed TS 5.6.4, "Monthly Operating Reports" and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. This is a Ginna TS Category (i) change.
- vi. TS 3.15 - The Applicability was revised to specify that LTOP is only required in MODES 5 and 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position. This change is consistent with the current requirements for isolating the SI pumps for LTOP conditions (3.3.1.7 and 3.3.1.8) and the ITS such that there is no real change in the MODE of Applicability. This is a Ginna TS Category (v.c) change.

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26. Technical Specification 3.16

- i. TS 3.16.1 and Table 3.16-1 - The requirements for the radiological environmental program which provides measurements of radiation and of radioactive materials in those exposure pathways and for specified radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- ii. TS 3.16.2 - The requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iii. TS 3.16.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

27. Technical Specification 4.0

- i. A new section SR 3.0.1 was added which establishes the requirements and limitations that the SRs must meet during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with SR 3.0.1. This SR provides clarifying and descriptive information for the SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.a) change.

- ii. TS 4.0 - This was revised to clarify the basic application of the 25% extension to routine surveillances consistent with the use and format of the ITS. The interval extension concept is based on scheduling flexibility for repetitive performances. There are clarifications provided in SR 3.0.2 for Surveillances which are not repetitive in nature and essentially have no interval as measured from the previous performance. This precludes the ability to extend these performances. The existing Specification 4.0 can be interpreted to allow the extension to apply to all Surveillances. An additional clarification provides the basis for consistency in scheduling flexibility for all performances of periodic requirements, whether they are Surveillances or Required Actions. The intent remains to perform the activity, on the average, once during each specified interval. This section does not provide any new requirements but provides clarification and a description of SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.

- iii. A new section SR 3.0.3 was added which establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. The SR permits the declaration of the LCO-not-met to be delayed for up to 24 hours or up to the limit of the specified Frequency (whichever is less), and eliminates confusion in applying the correct ACTION time limits at the end of this 24 hour period. The vast majority of surveillances performed demonstrate that systems or components, in fact, are OPERABLE. When a Surveillance is missed, it is primarily a question of OPERABILITY that has not been verified by the performance of the required surveillance. Based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance, the NRC has concluded that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times of the ACTIONS are less than the 24 hour limit or a shutdown is required to comply with ACTIONS (Ref. 53). This section, in part, provides new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.

- iv. A new section SR 3.0.4 was added which establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with SR 3.0.4. This SR provides clarifying and descriptive information for the SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.a) change.

28. Technical Specification 4.1

- i. The following changes were made to TS 4.1.1 or Table 4.1-1:

a. Table 4.1-1, Columns 2 (Calibrate) and 3 (Test) - Various calibration and testing interval requirements for RTS and ESFAS Functions were revised consistent with NUREG-1431. Changes to the testing interval requirements different from those identified and discussed in NUREG-1431 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (v.b.15) change.

b. The following new requirements were added to Table 4.1-1 (Ginna TS Category (iv.a) changes):

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1. ~~SR 3.4.2.13~~ ~~4.2.2~~ - requires verification every 30 minutes that T_{avg} for each RCS loop is $> 540^{\circ}\text{F}$ when any RCS loop T_{avg} is known to be $< 547^{\circ}\text{F}$: This surveillance is intended to ensure that the minimum temperature for criticality is not exceeded when the RCS is at less than Hot Zero Power conditions (i.e., 547°F). The surveillance is not required to be performed if the low T_{avg} alarm in each loop is reset with a setpoint $> 540^{\circ}\text{F}$.
2. SR 3.4.3.1 - requires verification every 30 minutes that RCS pressure, temperature, heatup and cooldown rates are within limits. This surveillance is only required during RCS heatup and cooldown operations, and inservice leak and hydrostatic testing. The 30 minute Frequency is based on the fact that heatup and cooldown rates are specified in hourly increments which provides adequate margin to correct minor deviations.

3. SR 3.4.1.1 - requires verification every 12 hours that pressurizer pressure is within limits during MODE 1. This surveillance is similar to current Ginna TS Table 4.1-1, #7 which is performed to support reactor trip functions.
4. SR 3.4.1.2 - requires verification every 12 hours that RCS average temperature is within limits during MODE 1. This surveillance is similar to current Ginna TS Table 4.1-1, #33 which is performed to support reactor trip functions.
5. SR 3.4.1.3 - requires performance of a precision heat balance to verify that RCS flow is within limits every 24 months. This surveillance is required to be performed within 7 days of entering MODE 1 and reaching 95% RTP.
6. SR 3.1.6.1 - Requires verification within 4 hours prior to criticality that the critical control bank position is within limits in the COLR.
7. SR 3.1.6.4 - Requires verification every 12 hours when critical that the sequence and overlap limits for the control banks not fully withdrawn are within limits specified in the COLR.
8. SR ~~3.1.8.43~~ 1.8.3 - Requires verification every 30 minutes during MODE 2 PHYSICS TESTS that THERMAL POWER \leq 5% RTP. Verification of the THERMAL POWER level will ensure that the initial conditions of the safety analyses are not violated.
9. SR 3.2.4.1 - Verification with a calculation using the power range channels every 7 days that the QPTR is within limits.

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SR 3.4.2.1 - requires verification within 30 minutes prior to achieving criticality that T_{min} for each RCS loop is $> 540^{\circ}F$. This surveillance is intended to ensure that the minimum temperature for criticality is not exceeded just prior to achieving criticality.

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- c. Table 4.1-1, Functional Units #1, #2, #3, #8, #17, #23, #25, #38a, #38b, #39, #40, #41a, and #41b - The notes or remarks which describe an operational detail, were not added. These details were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.

- d. LCO 3.3.1, Table 3.3.1-1, Function #10 was added for the RCP Breaker Position. This function anticipates the Reactor Coolant Flow - Low trips by monitoring each RCP breaker position to avoid RCS heatup that would occur before the low flow trip actuates. The function ensures that protection is provided against violating the DNBR limit due to loss of flow in either a single loop or two loop configuration. This is a Ginna TS Category (iv.a) change.
- e. LCO 3.3.1, Table 3.3.1-1, Function #14 was added for the SI Input from ESFAS. This function ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. A reactor trip is initiated every time an SI signal is present. This is a Ginna TS Category (iv.a) change.
- f. SR 3.3.1.14, SR 3.3.1.15, SR 3.3.1.16, SR 3.3.1.17, SR 3.3.1.18 were added for the Reactor Trip System Interlocks (P-6 through P-10). These surveillances are provided to ensure reactor trips are in the correct configuration for the current plant status. They are provided to back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Functions are not bypassed. This is a Ginna TS Category (iv.a) change.
- g. Table 4.1-1, Functions #34 and #35 - The requirements for the chlorine gas and ammonia gas instrumentation monitors for control room habitability were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

- h. Table 4.1-1, Functional Units #1 and 2 were revised to require a CHANNEL OPERATIONAL TEST (COT) on the power range and the intermediate range channels within 7 days prior to reactor criticality. The ITS Bases states that the 7 day time limits is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating the PHYSICS TESTS. This is a Ginna TS Category (iv.a) change.
- i. Table 4.1-1, Functional Unit #4 was revised to include a note requiring a channel check every 30 minutes while implementing MODE 2 PHYSICS TEST exceptions. Verification of the RCS temperature will ensure that the initial conditions of the safety analyses are not violated. This is a Ginna TS Category (iv.a) change.
- j. Table 4.1-1, Functional Units #18, #28, and #29 - The Surveillance requirements for radiation monitors R-1 through R-9 and R-17, emergency plan radiation instruments, and environmental monitors, were not added to the new specifications. These process variables are not an initial condition of a DBA or transient analysis. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- k. (47) ~~Table 4.1-1, Functional Unit #25 - The calibration and testing requirements for the containment pressure narrow range transmitter were not added to the new specifications. This instrument is not used or credited in any DBA or transient analysis. This instrument is only used to verify that containment pressure remains ≤ 1.0 psig and ≥ 2.0 psig during normal operation. These items were relocated to the TRM. This is a Ginna TS Category (iii) change.~~
- l. Table 4.1-1, Functional Unit #3 - This was revised to add a requirement which establishes a surveillance for a SRM CHANNEL CALIBRATION in MODE 6. This calibration consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data and is consistent with current Ginna Station procedures. This is a Ginna TS Category (iv.a) change.
- m. Table 4.1-1, Functional Units #14, #16, and #19 were relocated to the TRM for the same reasons as described in Section D, items 12.i through 12.iv.

These are Ginna TS Category (iii) changes.

- ii. The following changes were made to TS 4.1.2 or Table 4.1-2:
- a. Table 4.1-2, #6a was revised to extend the surveillance Frequency of the control rod exercises from monthly to every 92 days. The ITS Bases states that the 92 day Frequency takes into consideration the other information available to the operator in the control room and the channel check which is performed more frequently and adds to the determination of rod operability. This is a Ginna TS Category (v.b.29) change.
 - b. Table 4.1-2, #5 and #6b were revised to remove reference to once every 18 months or each refueling shutdown from the Frequency. These surveillances are only performed during a plant outage or during plant startup, prior to reactor criticality after each removal of the reactor head. This is a Ginna TS Category (v.c) change.
 - c. Table 4.1-2, Functional Unit #7 was revised to relocate the surveillance Frequency of the pressurizer safety valves to the IST Program consistent with SR 3.4.10.1. The Frequency continues to remain in a program requiring NRC approval. This is a Ginna TS Category (iii) change.
 - d. Table 4.1-2, Functional Unit #10 was not added to the new specifications. The requirement for verifying the refueling system interlocks is not an initial condition of a DBA or transient analysis. This requirement does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This change is consistent with WCAP-11618 (Ref. 52) and is a Ginna TS Category (iii) change.
 - e. Table 4.1-2, Functional Unit #13 was revised per SR 3.6.6.8 to require verification of the spray additive tank NaOH concentration once every 184 days instead of monthly. This change is acceptable since the spray additive tank is normally maintained isolated at power such that changes to the NaOH concentration or level are not expected. This is a Ginna TS Category (v.b.30) change.

- f. Table 4.1-2, Functional Unit #15 was revised to require RCS water inventory balances every 72 hours during steady state operation versus daily consistent with SR 3.4.13.1. This increased surveillance interval is considered acceptable based on the leakage detection systems required to be OPERABLE by LCO 3.4.15 and the various indications available to operators (e.g., volume control tank level and radiation alarms). This is a Ginna TS Category (v.b.31) change.
- g. Table 4.1-2, Functional Unit #17 was revised to only require verification of SFP boron concentration once every 31 days when fuel is stored in the SFP and the position of fuel assemblies which were moved in the SFP have not been verified. The current monthly requirement (regardless of the status of the SFP verification) is not reflected in the fuel handling accident analysis which does not credit the availability of soluble boron. This is a Ginna TS Category (v.b.32) change.
- h. Table 4.1-2, Functional Unit #19 - The trip function requirement for the Circulation Water Flood Protection was not added. The Circulation Water Flood Protection instruments only provide an anticipatory turbine trip and is not assumed in the Ginna Station safety analysis. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- i. The following new requirements were added to Table 4.1-2 (Ginna TS Category (iv.a) changes):
1. SR 3.1.1.1 - Requires verification every 4824 hours that the SHUTDOWN MARGIN is within the limits. The ITS Bases state that a Frequency of every 4824 hours is based on the generally slow change in boron concentration and the low probability of an event occurring without the required SDM.

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2. SR 3.1.3.1 - Requires verification prior to entering MODE 1 after each refueling that MTC is within the upper limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.
3. SR 3.1.3.2 - Requires verification prior to entering MODE 1 after each refueling that MTC will be within the 70% RTP MTC upper limit and the EOL lower MTC limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.
4. SR 3.1.8.3 - Require verification every 24 hours while implementing the MODE 2 PHYSICS TESTS exceptions that the SHUTDOWN MARGIN is within the limits. The ITS Bases state that a Frequency of every 24 hours is based on the generally slow change in boron concentration and the low probability of an event occurring without the required SDM.
5. SR 3.5.1.1 - requires verification every 12 hours that each accumulator motor-operated isolation valve is fully open above 1600 psig.
6. SR 3.5.1.3 - requires verification every 12 hours of an upper limit for the nitrogen pressure blanket in the accumulators to prevent lifting of the relief valve and overpressurization of the tank. A value of 790 psig was selected since it is above the accumulator pressure upper alarm setpoint of 760 psig and below the relief valve setpoint of 800 psig.
7. SR 3.5.1.4 - requires verification every 31 days on an STAGGERED TEST BASIS of an upper limit for boron concentration in the accumulator since this limit is used in determining the time frame which boron precipitation is addressed post LOCA. The value specified in the COLR was selected since this would not create the potential for boron precipitation in the accumulator assuming a containment (and accumulator) temperature of 60°F. This is also bounded by the containment sump pH calculations and assumptions used for chemical spray effects.

8. SR 3.5.1.5 - requires verification every 31 days that power is removed from the accumulator isolation valve operator above 1600 psig. This surveillance is consistent with current TS 3.3.1.1.i. A value of 1600 psig was selected (i.e., the same value as that for accumulator operability) since the RCS pressure interlock (i.e., P-11) as discussed in NUREG-1431 does not exist at Ginna Station. Therefore, there is no interlock signal to open the isolation valves in the event that they are closed.
9. SR 3.5.4.2 - requires verification every 7 days of an upper limit for boron concentration in the RWST since this limit is used in determining the time frame which boron precipitation is addressed post LOCA. The value specified in the COLR was selected since this would not create the potential for boron precipitation in the RWST assuming an Auxiliary Building (and RWST) temperature of 50°F. This is also bounded by the containment sump pH calculations and assumptions used for chemical spray effects.
10. SR 3.6.5.1 - requires verification every 24¹² hours that containment average air temperature is $\leq 120^\circ\text{F}$.
11. SR ~~3.6.6.73~~ ~~6.6.8~~ - requires verification every 184 days that the spray additive tank volume is ≥ 4500 gallons.
12. SR 3.7.11.1 - requires verification every 31⁷ days that ≥ 23 feet of water is available above the top of the irradiated fuel assemblies seated in the storage racks during fuel movement in the SFP. This verification is required since the fuel handling accident assumes that at least 23 feet of water is available with respect to iodine releases.
13. SR 3.7.13.1 and SR 3.7.13.2 - verification prior to fuel movement in the SFP that the associated fuel assembly meets the necessary requirements for storage in the intended region (e.g, enrichment limit, burnable poisons present). This verification is required to limit the amount of time that a fuel assembly could be misloaded in the SFP.

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14. SR 3.7.6.1 - requires verification every 12 hours that the CST volume is $\geq 22,500$ gallons. This ensures that the minimum volume of condensate is available for the preferred AFW System following an accident.
15. SR 3.7.7.1 - requires verification every 31 days that each CCW manual and power operated valve in the CCW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the CCW System is capable of performing its function following a DBA to provide cooling water to safety related components.
16. SR 3.7.7.2 - requires performance of a complete cycle of each CCW motor operated isolation valve to the RHR heat exchangers in accordance with the IST Program. This ensures that the normally closed motor operated valves are capable of being opened following a DBA.
- 142 17. SR ~~3.7.8.13~~ 7.8.2 - requires verification every 31 days that each SW manual and power operated valve in the SW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the SW System is capable of performing its function following a DBA to provide cooling water to safety related components.

1718. SR ~~3.4.15.53~~ 9.4.1 and 3.9.5.1 - requires a CHANNEL CALIBRATION of the containment air cooler condensate system monitor verification every 24 months when this system is in MODE 6 that one RHR loop is being used in place of the containment sump monitor in operation and circulating reactor coolant.

89 This ensures that the RCS is being mixed as assumed for boron dilution events and that decay heat removal continues during shutdown.

191 19. SR 3.9.3.1 - requires verification every 7 days that all containment penetrations which communicate to the outside environment are in

their required state in MODE 6. This ensures that containment is in the correct state prior to and during fuel movement.

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20. SR 3.6.3.1 - requires verification every 31 days that the mini-purge valves are closed, except when the penetration flow path is being used under administrative control. This ensures that the flow paths which provide a direct path from containment to the outside environment are in the correct position.

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21. SR 3.7.8.1 - requires verification every 24 hours that the screenhouse bay water level and temperature are within limits. This ensures that the ultimate heat sink source is within the assumptions of the accident analyses.

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22. SR 3.7.8.3 - requires verification every 31 days that all SW loop header cross-tie valves are in the correct position. This ensures that the valves are either locked opened or closed as necessary to support the accident analyses.

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23. SR 3.5.2.7 - requires a visual verification every 24 months that the RHR containment sump suction inlet line is not obstructed and that the screen shows no evidence of structural distress or abnormal corrosion. This ensures that the RHR system will not become plugged by expected debris which may exist in containment post-LOCA.

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24. SR 3.1.3.3 - requires verification prior to entering MODE 1 after each refueling that MTC will be within the EOL lower MTC limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

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25. SR 3.2.3.1 - requires verification every 12 hours that the AFD monitor is OPERABLE. This ensures that the AFD monitor is available to detect changes in AFD and provide necessary indication to operators.

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26. SR 3.2.4.1 - requires verification every 12 hours that the QPTR monitor is OPERABLE. This ensures that the QPTR monitor is available to detect changes in QPTR and provide necessary indication to operators.

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27. SR 3.6.6.5 - requires verification every 31 days that cooling water is flowing through each CRFC

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unit. This ensures that the heat removal capability of the CRFC units is verified to be available as assumed in the accident analyses.

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28. SR 3.5.2.3 - requires verification every 31 days that each breaker or key switch is in the correct position of valves required to be depowered or powered. This ensures that no single active failure will fail both ECCS trains.

- j. Table 4.1-2, Functional Units #1 and #2 - These were not added to the new specifications for the reasons discussed in Section D, item 11.i. This is a Ginna TS Category (iii) change.
- k. Table 4.1-2, Functional Unit #16 - This was revised to only require a verification of DG fuel oil inventory once every 31 days instead of daily. Since the storage tanks are of passive design and are provided with various level alarms, verification every 31 days is considered adequate. This is a Ginna TS Category (v.b.33) change.

1. Table 4.1-2, Functional Unit #4 - This was relocated to the TRM for the same reasons as described in Section D, item 12.iv. This is a Ginna TS Category (iii) change.

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m. Table 4.1-2, Functional Unit #12 - This was relocated to the TRMIST Program since it does not meet any of the requirements for inclusion in the ITS. This is a Ginna TS Category (iii) change.

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n. Table 4.1-2, Functional Unit #9~~18~~ - The Frequency for determining gross specific activity of the secondary system was revised from once every 72 hours to once every 31 days. In addition, the determination of I-131 was also changed to once every 31 days independent of the last activity level since the current Ginna TS allow up to 6 months between tests. These changes are all consistent with NUREG-1431. This is a Ginna TS Category (v.e) ~~change~~ ~~510~~ change.

iii. The following changes were made to TS 4.1.3 or Table 4.1-3:

a. Table 4.1-3 - The Post Accident Monitoring Instrumentation Functions required by this specification were revised to include only Regulatory Guide 1.97, Type A and Category 1 variables. These Functions are denoted in UFSAR Table 7.5-1 and have been previously reviewed and approved by the NRC (Ref. 35). This is a Ginna TS Category (v.c) change.

iv. The following changes were made to TS 4.1.2 or Table 4.1-4:

- a. Table 4.1-4, Functional Unit #1 was revised per SR 3.4.16.1 to only require verification of reactor coolant gross specific activity once every 7 days when $T_{avg} \geq 500^{\circ}F$ versus once every 72 hours above Cold Shutdown (i.e., $T_{avg} \geq 200^{\circ}F$). The increased surveillance interval is acceptable based on the small probability of a gross fuel failure during the additional 4 days. Fuel failures are more likely to occur during startup or fast power changes and not during steady state power operation during which the majority of sampling is performed. Gross fuel failures will also result in Letdown radiation alarms and possibly containment radiation alarms providing additional operator indication. Only requiring this surveillance when $T_{avg} \geq 500^{\circ}F$ provides consistency with the LCO Applicability. This is a Ginna TS Category (v.b.34) change.

b. Table 4.1-4, Functional Unit #2 was revised per SR 3.4.16.2 to require verification of DOSE EQUIVALENT I-131 when $T_{avg} \geq 500^\circ\text{F}$ instead of above 5% reactor power. This conservative change provides consistency with the LCO Applicability. This is a Ginna TS Category (v.a) change.

c. Table 4.1-4, Functional Unit #3 was revised per SR 3.4.16.3 to delay determination of E until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation following the reactor being subcritical for ≥ 48 hours. The 31 days was added to ensure that radioactive materials are at equilibrium in order to provide a true representative sample for E determination and eliminate possible false samples. This is a Ginna TS Category (v.e) ~~change. 53) change.~~

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v. The following changes were made to TS 4.1.4 or Table 4.1-5:

a. Table 4.1-5, Functional Unit #3b was revised to require a channel check of particulate sampler R-11 every 12 hours versus weekly. This is required since R-11 is being used to monitor RCS leakage and may be the only installed system OPERABLE to perform this task for up to 30 days per new LCO 3.4.15.

b. TS 4.1.4 and Table 4.1-5 - The Radioactive Effluent Monitoring Instrument Functions required by this specification were not added to the new specifications since these process variables are not an initial condition or a DBA or transient analysis. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and were relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

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c. TS 4.1-5, Functional Unit #3a and #3b were revised to only require the functional test of the valves actuated by R-11 and R-12 once every 24 months versus quarterly. This change is consistent with NUREG-1431 and is considered acceptable since these channels are redundant to the containment isolation signal. As such the accident analysis do not take specific credit for R-11 and R-12 to isolate the containment purge valves. Also, a functional test of the channels (minus actuation of the valves) is to be done quarterly. This is a Ginna TS Category (v.b.49) change.

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d. The CHANNEL CHECK of R-11 was revised from weekly to daily in MODES 1, 2, 3, and 4, and during MODE 6 when required by LCO 3.9.3. This is a conservative change which requires R-11 and R-12 to be checked at the same frequency. This is a Ginna TS Category (v.a) change.

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e. Table Note 5 was deleted since requiring that the CHANNEL CALIBRATION be traced back to the National Bureau of Standards is not a necessary level of detail to be contained in TS. NUREG-1431 does not contain this level of detail since it does not meet any of the four criteria. Therefore, this Note is relocated to plant procedures. This is a Ginna TS Category (iii) change.

29. Technical Specification 4.2

i. TS 4.2.1 - The specific requirements for the Inservice Inspection Program, which include Quality Groups A, B, and C components, high energy piping outside of containment, snubbers and steam generator tubes, were not added. The level of detail is relocated to licensee controlled documents (Ginna Station QA Manual, Appendix B) and a more generic description is provided. This is a Ginna TS Category (iii) change.

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TS 4.2.1 - The title of the "Ginna Station QA Manual" was changed to "Nuclear Policy Manual" since this is the location of the IST Program per Reference 64. This is a Ginna TS Category (vi) change.

30. Technical Specification 4.3

- i. TS 4.3.5.6 - This surveillance was not added for the reasons discussed under Section C, item 6.xi. This Surveillance has been relocated to the TRM. This is a Ginna TS Category (iii) change.
- ii. TS 4.3.5.3.b - This surveillance was not added since performance of pump testing in accordance with the Inservice Testing program should not be required for an operating RHR pump. The status of a non-operating RHR pump is assured by new SR 3.4.6.3 which requires the verification of the breaker alignment and indicated power available to the pump. The Inservice Testing program testing is mainly performed to ensure adequate performance during accident conditions which far exceeds the requirements during shutdown conditions. This test is not necessary to ensure operability during MODE 4 operations. However, this Surveillance is required for ECCS during MODE 4 (see new SR 3.5.3.1) This is a Ginna TS Category (v.c) change.
- iii. TS 4.3 - The following new requirements were added (Ginna TS Category (iv.a) change):
 - a. SR 3.4.6.3, 3.4.7.3 and 3.4.8.2 - Requires the verification of correct breaker alignment for the non-operating, but required, RHR pump in MODES 4 and 5.
 - b. SR 3.4.9.2 - Requires verification that the total capacity of the pressurizer heaters is ≥ 100 KW once every 92 days.
 - c. SR 3.4.11.2 - Requires a complete cycle of each PORV using the nitrogen system once every 24 months.

- iv. TS 4.3.3.1, 4.3.3.2, and 4.3.3.3 - The requirement that the leakage tests be performed with a minimum test differential pressure of 150 psid was not added to the new specifications. The bases for new LCO 3.4.14 reference ASME, Section XI (Ref. 53) which provides acceptable guidance for performing these leakage tests. This includes adjusting the observed leakage rates for tests that are not conducted at the maximum differential pressure by assuming that leakage is directly proportional to the pressure differential to the one half power. This is a conservative change in most cases since it requires that the PIVs be tested under the maximum differential pressure conditions. This is a Ginna TS Category (v.c) change.

v. TS 4.3.3.4 - The allowed leakage rates for PIVs was adjusted from a single value for all valves to a value based on valve size consistent with SR 3.4.14.1 and SR 3.4.14.2. This change provides greater information of valve degradation and removes an unjustified penalty on larger valves (Ref. 54). This is a Ginna TS Category (v.c) change.

vi. TS 4.3.5.5 - This surveillance was not added during MODE 1 operation since there is a reactor trip function which protects the SG level. This is a Ginna TS Category (i) change.

vii. TS 4.3.1.1 - This requirement was not added to the new specifications since it only states that the reactor vessel must be tested in accordance with 10 CFR 50, Appendix H. Since this requirement is already specified in the CFR, it does not have to be retained with the TS and was deleted. This is a Ginna TS Category (ii) change.

viii. TS 4.3.3.1 - This was modified to remove the requirement to test the SI cold leg injection and RHR RCS PIVs each cold shutdown. At Ginna Station, these flowpaths are only used for emergency injection (i.e., they are not relied upon or used during cold shutdown conditions). Since the valves are maintained closed at all times, requiring a leak test within 24 hours of being opened or having maintenance performed, and once every 24 months provides adequate protection. A leakage test every 24 months is also consistent with NRC approved OMa-1988. This is a Ginna TS Category (v.b.35) change.

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TS 4.3.4.2 - This was revised to limit the exclusion for testing of the PORV block valves from when "the valve is closed," to "when the valve is closed due to PORV leakage > 10 gpm." This ensures that the block valve is tested under all conditions except those that could potentially result in a plant transient. This is a conservative change. This is a Ginna TS Category (v.a) change.

31. Technical Specification 4.4

i. TS 4.4.4 - The requirements for the tendon stress surveillances were not added. The level of detail is relocated to the Pre-stressed Concrete Containment Tendon Surveillance Program described in new Specification 5.5.6 and a more generic program description is provided. This is a Ginna TS Category (iii) change.

ii. TS 4.4.3 - The requirements for the testing of the portion of the RHR system in the recirculation configuration were not added. The level of detail is relocated to the Primary Coolant Sources Outside Containment Program described in new Specification 5.5.2 and a more generic program description is provided. This is a Ginna TS Category (iii) change.

- iii. TS 4.4.1 (except definition for L_a), 4.4.2.1, 4.4.2.2, and 4.4.2.4 - These were not added to the new specifications since this information is contained in 10 CFR 50, Appendix J and does not need to be retained within technical specifications. SRs 3.6.1.1 and 3.6.1.2 provide for the necessary relation from technical specifications to Appendix J (see also Reference 63). These are Ginna TS Category (ii) changes.
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- iv. TS 4.4.2.3.a and 4.4.2.3.b - These were revised to require that if the allowed 10 CFR 50, Appendix J leakage limits are exceeded, they must be restored within 1 hour versus 48 hours consistent with LCO 3.6.1. However, the leakage limit of $< 0.6 L_a$ was revised to be consistent with the new Appendix J rule and implementation guidance (i.e., the leakage limit is $< 0.6 L_a$ on a maximum pathway leakage rate basis prior to entering MODE 4 for the first time following each refueling outage and $< 0.6 L_a$ on a minimum pathway leakage rate basis for all other time periods) (see also Reference 63). This is a Ginna TS Category (v.a) change.
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- v. TS 4.4.2.4.c - A specified air lock leakage acceptance criteria of $\leq 0.05L_a$ when tested at $\geq P_a$ was added to the new specifications. This acceptance criteria is required to be retained within technical specifications by 10 CFR 50, Appendix J, Section III.D.2(iv) and is consistent with NUREG-1431 and current testing requirements. In addition, a new Surveillance was added to verify that only one door in each airlock can be opened at a time once every 24 months. This test is necessary to ensure that the OPERABILITY of the airlocks, as defined in the new bases for LCO 3.6.2 is maintained. These are Ginna Category (iv.a) changes.

- vi. TS 4.4.2.3.c - The requirement to perform an engineering evaluation if the mini-purge supply and exhaust lines isolation valve leakage exceeds 0.05 L_a was revised to require isolation of the affected penetration within 24 hours. In addition, the affected penetration must be verified isolated once every 31 days if it is outside containment, or once every 92 days if it is inside containment. These changes provide direct guidance to operators which are consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.
- vii. ⁽¹²⁶⁾ TS 4.4.5.1 - Two new surveillances (SR ~~3.6.3.1~~ ~~3.6.3.2~~ and SR ~~3.6.3.2~~ ~~3.6.3.3~~) were added which require verification of the correct position of containment isolation barriers located outside containment once every ~~184~~ 92 days and inside containment prior to entering MODE 4 from MODE 5 if it has not been performed within the previous ~~184~~ 92 days. These surveillances ensure that the containment isolation barriers remain OPERABLE above MODE 5. These are Ginna TS Category (iv.a) changes. ⁽¹²²⁾
- viii. TS 4.4.6.2 - The Surveillance Frequency for automatic containment isolation valves has been revised from 18 to 24 months (see Section D, item 1.xii). The response times for CIVs is discussed in the bases for new LCO 3.6.3. This is a Ginna TS Category (v.b.1) change.
- ix. ⁽⁴⁵⁾ TS 4.4 - Two new Surveillances were added with respect to the hydrogen recombiners (SR 3.6.7.1 and SR 3.6.7.2). The first new Surveillance requires ~~that the blower fan for a functional check of the hydrogen recombiners be operated for ≥ 5 minutes~~ once every 24 months. The second new Surveillance requires that a CHANNEL CALIBRATION be performed on the hydrogen recombiner actuation and control channels once every 24 months. The performance of these SRs ensures that the hydrogen recombiners are OPERABLE and capable of performing their post-accident function. These are Ginna TS Category (iv.a) changes.
- x. TS 4.4.7 - The Frequency for performance of a CHANNEL CHECK of the hydrogen monitors was revised from daily to monthly. In addition, the Frequency for CHANNEL CALIBRATIONS was revised from quarterly to every 24 months. These changes are consistent with NUREG-1431 and are justified by industry experience. These are Ginna TS Category (v.b.21) changes.

32. Technical Specification 4.5

- i. TS 4.5.1.1.a - This was revised to delete the statement that the SI and RHR pumps are prevented from starting during this test. Since these components have recirculation lines available, this statement is not required. This is a Ginna TS Category (v.c) change.
- ii. TS 4.5.2.1 - This was revised to relocate all SI, RHR, and CS pump testing frequencies and discharge pressure requirements to the Inservice Testing program described in new Specification 5.5.8 consistent with the ITS. These are Ginna TS Category (iii) changes, respectively.
- iii. TS 4.5.2.2.c - The test related to accumulator check valve testing for operability every refueling shutdown was relocated to the Ginna Station Inservice Testing program. The valves are currently partially stroke tested quarterly and refurbished every six years. Leakage associated with these check valves is addressed by SR 3.5.1.2. This is a Ginna TS Category (iii) change.
- iv. The following new ITS testing requirements were added (Ginna TS Category (iv.a) change):

- a. SR 3.5.2.1 - requires verification every 12 hours that ECCS related isolation valves are in their required position. These valves are currently specified in TS 3.3.1.1.g, 3.3.1.1.i, and 3.3.1.1.j.
 - b. SR 3.5.2.2 - requires verification every 31 days that ECCS related valves which are not locked, sealed, or otherwise secured in position are in their correct position.
- v. TS 4.5.2.3 - The requirements denoting the Frequency and conditions of the air filtration system tests were not added to the new specifications. This level of detail is relocated to the Ventilation Filter Testing Program described in new Specification 5.5.10. In addition, the remaining requirements were all relocated to the Administrative Controls section. These are Ginna TS Category (iii) and (i) changes, respectively.
- vi. TS 4.5.2.3.6.a - These test requirements were revised to clarify that two separate tests are performed. A HEPA filter test and a charcoal adsorber bank test are separately performed with each requiring a limit of less than 3 inches of water. This is essentially equivalent to a combined test of less than 6 inches of water and is consistent with specified testing standards. This is a Ginna TS Category (vi) change.

- vii. 43 TS 4.5.1.2 - ~~Two~~ new Surveillance Surveillances (SR ~~3.6.6.1~~ and SR ~~3.6.6.2~~) were added to verify the correct position of each manual, power operated, and automatic valve in the NaOH and CS flowpath that is not locked, sealed, or otherwise secured in position. This Surveillance ensures that the NaOH and CS System ~~is~~ Systems are OPERABLE in accordance with the LCO. ~~This is a~~ These are Ginna TS Category (iv.a) ~~change~~ changes.
- viii. TS 4.5.1.2.b - The Frequency of performing the spray nozzle gas test was revised from once every 5 years to once every 10 years consistent with SR 3.6.6.14. The increased surveillance interval is considered acceptable due to the passive nature of the spray nozzles and previous acceptable results. This is a Ginna TS Category (v.b.36) change.
- ix. 124 TS 4.5.2.3.5 - This was revised to only require actuation of the post-accident charcoal filter dampers from an actual or simulated SI signal once every 24 months to ensure that the system aligns itself correctly (SR ~~3.6.6.12~~ ~~3.6.6.15~~). The post-accident charcoal filter dampers must still be opened at least once per 31 days to allow the system to operate for ≥ 15 minutes. Consequently, only the frequency of the automatic alignment of the dampers is being revised to provide consistency with other specifications. This is a Ginna TS Category (v.b.37) change.
- x. 123 TS 4.5.2.2.a - This was revised to adjust the testing Frequency of the spray additive valves from monthly to once every 24 months consistent with SR ~~3.6.6.13~~ ~~3.6.6.16~~. This increased testing interval is acceptable since the system only needs to be verified that it can actuate on an actual or simulated SI signal on a refueling basis similar to the SI and RHR systems. Any additional valve testing is addressed by the IST program. In addition, a new Surveillance (SR ~~3.6.6.9~~ ~~3.6.6.12~~) was added to verify that the CS motor operated isolation valves actuate to their correct position once every 24 months following an actual or simulated SI signal. Finally, a new Surveillance (SR ~~3.6.6.14~~ ~~3.6.6.17~~) was added to verify that the spray additive flow rate is within limits once every 5 years. These changes ensure that the CS and spray additive tank LCOs continue to be met. These are Ginna TS Category (v.b.38) changes.
- xi. 123 TS 4.5.2.3.3 and 4.5.2.3.4 - These were revised to require that each CRFC unit be operated for ≥ 15 minutes once every 31 days (SR ~~3.6.6.2~~ ~~3.6.6.4~~). This test will ensure that the CRFC units are OPERABLE in accordance with the LCO. In addition, a new Surveillance is also required once every 24 months to ensure that the CRFC units start on an actual or simulated SI signal. These tests will ensure that the CRFC units are OPERABLE in accordance with the LCO. These are Ginna TS Category (v.a) changes.

- xii. TS 4.5.2.3.9 - This was revised to require a test of the automatic actuation capability of the CREATS once every 24 months. This verification is necessary to ensure that the control room environment can be isolated in the event of a radiological release. This is a Ginna TS Category (iv.a) change.

33. Technical Specification 4.6

- i. TS 4.6.1.a - The cold or refueling requirements (MODES 5 and 6) for demonstrating DG operability have been revised to include (1) verification of DG day tank fuel oil level, (2) verification of the onsite supply of fuel oil, and (3) operation of the fuel oil transfer system. These are consistent with the required surveillances for DG operability in MODES 1, 2, 3, and 4 and provide assurance that the DG is OPERABLE. This is a Ginna TS Category (iv.a) change.
- ii. TS 4.6.1.b.6 - The requirement to verify that the DG is aligned to provide standby power to the associated emergency buses was not added. This requirement is within the definition of an OPERABLE DG and is denoted in the bases of new TS 3.8.1. This is a Ginna TS Category (i) change.

iii. TS 4.6.1.c - The requirement to perform the tests in Specification 4.6.1.b prior to exceeding cold shutdown was not added. This requirement was replaced with a general provision (new SR 3.0.4) that restricts entry into a MODE or other specified condition in the Applicability of an LCO unless the LCO's surveillances have been met. This is a Ginna TS Category (i) change.

iv. TS 4.6.1.d - The diesel fuel oil test requirements were relocated to new TS 5.5.12 and are proposed to be identified as a "program" consistent with the format of NUREG-1431. ~~This is a Ginna TS Category (i) change.~~ In addition, the fuel oil testing program was revised to expand the testing requirements consistent with NUREG-1431 and delete the 92 day test of the stored fuel oil.

(18c)

The fuel oil must now be tested before being placed in the storage tanks such that testing of viscosity, water, and sediment after being placed in the storage tanks is no longer required. This is a conservative change which reduces the potential to harm the safety related diesel generators from "bad" fuel oil. This is a Ginna TS Category (v.a) change.

- v. TS 4.6.1.e.1 - The requirement to inspect the DG in accordance with the manufacturer's recommendations was not added. No screening criteria apply for this requirement since DG inspections are not part of the primary success path assumed in the mitigation of a DBA or transient. The requirement does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- vi. TS 4.6.1.e.3(b) - The requirement for DG testing simulating a loss of offsite power in conjunction with a safety injection test signal was revised. Details of the test acceptance criteria were relocated to the ~~TRM~~ ~~and~~ ~~procedures~~ since this level of detail is not typically specified in the SR. This is a Ginna TS Category (iii) change.
- vii. TS 4.6.2.a and 4.6.2.b - The station battery testing requirements were revised to add acceptance criteria, parameters, and associated actions for battery operability supporting DC electrical power subsystems. These requirements are provided in the bases SRs and enhance the current criteria specified in the TS and is a conservative change regarding the definition of battery OPERABILITY. In addition, the electrolyte temperature is only to be measured every 92 days versus monthly consistent with IEEE-450 requirements. These are Ginna TS Category (iv.a) and (v.a) changes, respectively.
- viii. TS 4.6.2.f - The details denoting battery degradation were moved to the bases and were revised to include expected life parameters of the battery when compared to a capacity criteria of 100% of the manufacturer's rating. This criteria is used in conjunction with identifying when the surveillance Frequency must be increased and is consistent with ITS. These are Ginna TS Category (iii) and (v.a) changes, respectively.

ix. TS 4.6.2 - Two new surveillances (SR 3.8.4.1 and SR 3.8.5.1) were added which require verification every 317 days that the battery charging capability terminal voltage is ≥ 150 -amps 129 V of float voltage during operating and shutdown conditions. This surveillance ensures that the required battery charger remains capable of maintaining DC system loads and a float charge on the battery. This is a Ginna TS Category (iv.a) change.

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x. TS 4.6.2.c - The requirement for trending battery test data was not added to the new specifications since this is trending must be performed to meet the Frequency requirements for SR 3.8.6.2 and SR 3.8.4.3. This therefore, this requirement is a Ginna TS Category (i) change relocated to plant procedures. This is a Ginna TS Category (iii) change.

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X1

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TS 4.6.1.b.4 - this was revised to require that the one hour monthly DG run must be performed after successful performance of the monthly DG start (i.e., TS 4.6.1.b.4) or the refueling outage test (i.e., TS 4.6.1.e.4). This ensures that the DG is not being unnecessarily started for the performance of the one hour run. This change is consistent with current testing practices and NUREG-1431. This is a Ginna TS Category (iv.a) change.

X1

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TS 4.6.1.b.4, 4.6.1.e.2, and 4.6.1.e.3 - These were revised to add a note to the surveillance which specifically states that credit may be taken for unplanned events that satisfies these SRs. This is consistent with current operating practice and NUREG-1431 since if a loss of offsite power were to occur requiring a DG run, it should be able to satisfy the surveillance if it meets all of the testing requirements. This also prevents unnecessary tests of the DGs which can lead to potential degradation. This is a Ginna TS Category (v.c) change.

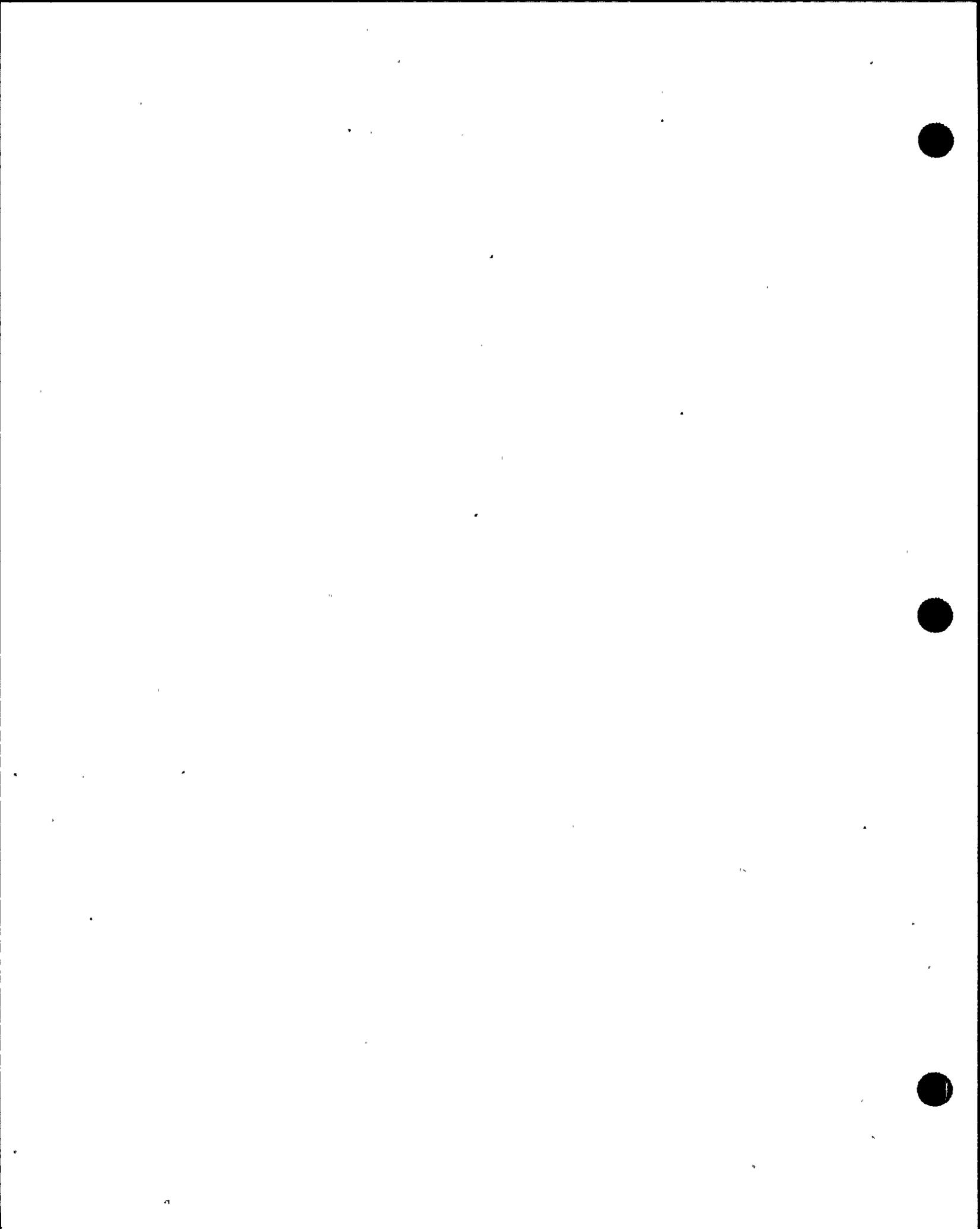
34. Technical Specification 4.7

- i. TS 4.7 was revised to include a surveillance to ensure that each MSIV can close on an actual or simulated actuation signal every 24 months consistent with NUREG-1431 and current Ginna Station TS Table 3.5-2 which require that the isolation signals to the MSIVs be OPERABLE. In addition, Required Actions were provided in the event that the MSIVs cannot close as required by this Surveillance. These actions require restoration of, or closure of an inoperable MSIV, within 24 hours. In the event that both MSIVs are inoperable, the plant must enter LCO 3.0.3. Finally, requirements for the main steam non-return check valves were added. These are Ginna TS Category (iv.a) changes.

35. Technical Specification 4.8

- i. TS 4.8.1 and 4.8.2 - The Frequency of the AFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.

- ii. TS 4.8.3 - This Surveillance was revised to relocate the Frequency of testing the AFW suction and discharge valves to the Inservice Testing Program which provides sufficient control of these testing activities. In addition, the cross-over motor operated isolation valves were not added to the new specifications since these valves are not credited in the accident analyses (see bases for new LCO 3.7.5). These are Ginna TS Category (iii) and (v.b.39) changes, respectively.



- iii. TS 4.8.4 - The Frequency of the SAFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.
 - iv. TS 4.8.5 - This Surveillance was revised to relocate the Frequency of testing the SAFW suction, discharge, and cross-over valves to the Inservice Testing Program which provides sufficient control of these testing activities consistent with NUREG-1431. This is a Ginna TS Category (iii) change.
 - v. TS 4.8.6 - This was revised to relocate the acceptance criteria for the AFW and SAFW tests to the actual procedures performing these tests. The new bases identify what is required for OPERABILITY of the AFW and SAFW Systems such that specifying this acceptance criteria is unnecessary. In addition, both the bases and test procedures are controlled under 10 CFR 50.59. This is a Ginna TS Category (iii) change.
 - vi. TS 4.8 - A new Surveillance was added requiring verification every 31 days of the correct position of each AFW and SAFW manual, power operated and automatic valve in the flow path that is not locked, sealed or otherwise secured in position. This verification is required to ensure that the AFW and SAFW Systems are OPERABLE when not in service. This is a Ginna TS Category (iv.a) change.
 - vii. TS 4.8.10 - The requirement to measure the response time of the AFW pumps and valves to be ≤ 10 minutes once every 18 months was not added to the new specifications. The time requirements for the AFW System are described in the new bases. While some accidents do not require AFW for 10 minutes, the small break LOCA and loss of feedwater transients require AFW within much shorter time frames. Therefore, this Surveillance is not accurate and is not required. This is a Ginna TS Category (v.b.40) change.

36. Technical Specification 4.9

- i. TS 4.9 - This was revised to include an LCO requirement that the measured core reactivity be within $1\% \Delta k/k$ of the predicted values and to add a specific surveillance Frequency of every 31 EFPD after the initial normalization. The Surveillance Requirement was divided into two surveillances to clarify the difference between the initial normalization and the monthly verification. These are Ginna TS Category (v.c) changes.

37. Technical Specification 4.10

- i. TS 4.10.1 and Table 4.10-1 - The requirements for the radiological environmental program which provides measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- ii. TS 4.10.2 - The requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iii. TS 4.10.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

38. Technical Specification 4.11

- i. TS 4.11.1 - The requirements denoting the Frequency and conditions of the SFP filtration system tests were not added. The level of detail is relocated to the VFTP described in new Specification 5.5.10. This is a Ginna TS Category (iii) change.
- ii. TS 4.11.1.1.a, 4.11.1.1.b, and 4.11.1.1.c - These charcoal adsorber system testing requirements were relocated to the VFTP described in the Administrative Controls (TS 5.5.10). This is a Ginna TS Category (i) change.
- iii. TS 4.11.1.1.d - This was not added to the new specifications since this verification is not required to ensure that initial assumptions of the accident analyses are still met. The SFP Charcoal Absorber System does not utilize heaters. The bases for SR 3.7.13.1 state that operating the ventilation system for ≥ 15 minutes every 31 days for systems without heaters is to ensure system operation. In accordance with new LCO 3.7.10 (NUREG-1431 LCO 3.7.13), the ABVS is required to be in operation during fuel movement within the Auxiliary Building. As such, the ABVS is not a standby system at Ginna Station (i.e., the system must be both OPERABLE and in operation during its MODE of Applicability). Therefore, a monthly verification provides no verification of any accident analysis assumption. Instead, ~~atwo~~ ~~new~~ ~~Surveillance~~ ~~was~~ ~~Surveillances~~ ~~were~~ ~~added~~ ~~which~~ ~~requires~~ ~~require~~ verification every 24 hours that the Auxiliary Building operating floor level is at a negative pressure with respect to the outside environment and that the ventilation system is in operation. ~~This verification is these~~ ~~verifications~~ ~~are~~ ~~consistent~~ ~~with~~ ~~plant~~ ~~practices~~ ~~and~~ ~~ensures~~ ~~that~~ ~~an~~ ~~initial~~ ~~assumption~~ ~~assumptions~~ ~~of~~ ~~the~~ ~~fuel~~ ~~handling~~ ~~accident~~ ~~is~~ ~~are~~ ~~being~~ ~~maintained~~. The change is also consistent with Reference 55. This is a Ginna TS Category (v.c) change.

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- iv. TS 4.11.2.1 - This was revised to only require verification of RHR pump OPERABILITY once every 12 hours versus 4 hours consistent with SR 3.9.3.1. A Frequency of 12 hours is adequate due to the alarms and indications available to the operators with respect to RHR pump and loop performance. This is a Ginna TS Category (v.b.41) change.
- v. TS 4.11.2.2 - This was revised to remove the requirement for an Inservice Test of the RHR pumps. An Inservice Test should not be required for an operating pump. The status of a non-operating RHR pump is assured by new SR 3.9.4.2 which requires the verification of the breaker alignment and indicated power available to the pump. The Inservice Testing program test is mainly performed to ensure adequate performance during accident conditions which far exceeds the requirements during normal conditions. This test is not necessary to ensure OPERABILITY during MODE 6 operations. This is a Ginna TS Category (v.b.42) change.
- vi. TS 4.11.3.1 -- This was revised to only require a verification of the water level in the reactor cavity within 24 hours of fuel movement versus 2 hours. The new TS usage rules state that a SR is to be continuously performed at its required Frequency. However, the SR is only required to be performed when in the MODE of Applicability. Therefore, a SR with a Frequency of 24 hours must have been performed within 24 hours before entering the MODE of Applicability. A Frequency of 24 hours is acceptable due to the large volume of water available and the procedural controls in place. This is a Ginna TS Category (v.c) change.

39. Technical Specification 4.12

- i. TS 4.12.1.1 and Table 4.12-1 - The requirements for radioactive material released in liquid effluents to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- ii. TS 4.12.1.2 - The requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, GDC 60 and 10 CFR Part 50, Appendix I, Section II.D, were not added. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iii. TS 4.12.2.1 and Table 4.12-2 - The requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iv. TS 4.12.2.2 - The requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- v. TS 4.12.3 - The requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

40. Technical Specification 4.13

- i. TS 4.13 - The requirements for periodic testing of leakage for radioactive sources were not added. The source leak test are not assumed in the analysis of any DBA or transient. Further, the leakage from radioactive sources is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ~~TRM~~ODCM. This is a Ginna TS Category (iii) change.

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41. Technical Specification 4.14

- i. TS 4.14 - The requirements for the testing of snubbers were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to Inservice Testing Program described in new Specification ~~5.5.85.5.7~~ and more generic program description is provided. This is a Ginna TS Category (iii) change.

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42. Technical Specification 4.15

None.

43. Technical Specification 4.16

- i. TS 4.16 - A new surveillance was added which requires verification ~~once within 12 hours and every 12 hours thereafter~~ that an accumulator's motor operated isolation valve is closed when its pressure is greater than or equal to the pressure allowed by the P/T limit curves provided in the PTLR consistent with SR 3.4.12.3. In addition, a verification ~~once within 12 hours and every 31 days thereafter~~ that power is removed to these isolation valves is also added. These verifications are needed to ensure that the accumulator does not discharge into the RCS and cause an overpressure event which challenges the LTOP System. This is a Ginna TS Category (iv.a) change.

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- ii. TS 4.16.1.a - This surveillance was revised to delay performance of the PORV functional channel test until 12 hours after decreasing to the LTOP enable temperature specified in the PTLR instead of within 31 days prior to entering the LTOP System Applicability. This change eliminates the performance of the functional test when RCS is between 330°F (the LTOP enable temperature) and 350°F (MODE 3 lower limit) during forced shutdowns. Instead, the test can be performed within 12 hours of entering the specified condition and reduces the immediate operator burden. This is a Ginna TS Category (v.b.43) change.

44. Technical Specification 5.1

- i. TS 5.1.1, TS 5.1.2, and Figure 5.1-1 - The description and figure of the site area boundary and ~~exclusion area boundary~~ was not added to the new specifications consistent with Traveller CEOG-03, C.1. Since the description of ~~these~~ ²²⁵ ~~this~~ design features ~~feature~~ does not satisfy ~~meet~~ the NRC Final Policy Statement ~~technical specification~~ screening criteria for Design Features described in 10 CFR 50.36, this description is relocated to licensee controlled documents (i.e., UFSAR, Section 2.1.2). ~~This is~~ ¹⁶⁹ ~~The~~ figure and description of the ~~exclusion area boundary~~ was also replaced with a Ginna TS Category ~~(iii)~~ ^{changeable} describing this feature consistent with Traveller CEOG-03, C.1. There are Ginna TS Category ~~(iii)~~ changes.

45. Technical Specification 5.2

- i. TS 5.2 - The description of the containment design features was not added. Specific containment features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). ~~Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria~~ ¹⁶⁹ ~~Therefore,~~ this description is relocated to licensee controlled documents (i.e., UFSAR Sections 3.8.1 and 6.2). This is a Ginna TS Category (iii) change.

46. Technical Specification 5.3

- i. TS 5.3.1.a and TS 5.3.1.c - The description of the reactor core design features was revised consistent with the standard guideline of NUREG-1431. The section now includes the amount, kind, and source of nuclear material related to the reactor core. This is a Ginna TS Category (v.c) change.
- ii. TS 5.3.1.b - The description of the fuel storage design feature with respect to the maximum enrichment weight percent was revised and relocated to new Specification 4.3.1. The changes are in accordance with the changes

discussed in item 47.ii, below. These are Ginna TS
Category (v.c) and (i) changes, respectively.

iii. TS 5.3.2 - The description of the reactor coolant system (RCS) design features was not added. Specific RCS features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). ~~Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria therefore,~~ this description is relocated to licensee controlled documents (i.e., UFSAR Section 3.7.1 and Chapter 5). This is a Ginna TS Category (iii) change.

iv. TS 5.3.1.b - This was revised to increase the fuel enrichment limit from 4.25 weight percent to 5.05 weight percent. This change has been evaluated and found to be acceptable with respect to postulated fuel handling accidents (Ref. 29). This is a Ginna TS Category (v.b.46) change.

47. Technical Specification 5.4

i. TS 5.4.1, 5.4.2, 5.4.6, and Figures 5.4-1 and 5.4-2 - The description of the fuel storage design features denoting spent fuel storage regions and borated water concentrations ~~was~~ were relocated to Chapters 3.7 and 3.9. These features are discussed in ~~LCO, LCO's~~ 3.7.11, ~~LCO~~-3.7.12, ~~LCO~~-3.7.13, and ~~LCO~~-3.9.1 as appropriate. In addition, appropriate Required Actions were added in the event that SFP water level, boron concentration, or SFP region storage requirements are not met. This is a Ginna TS Category (i) change.

ii. TS 5.4.2 - The description of the fuel storage design features was revised. The revision to these features are based on a revised criticality analysis supporting the proposed 18 month fuel cycle (Reference 29). The description of these features follow the standard guideline of NUREG-1431 which would include the amount, kind, and source of special nuclear material with the exception that nominal center to center spacing between the fuel assemblies was not added. This is a Ginna TS Category (v.c) change.

iii. TS 5.4.3 - The description of the fuel storage design feature denoting the 60-day limit on storage of discharged fuel assemblies in Region 2 was not added. No screening criteria applies for the time limit on storage of discharged fuel assemblies in Region 2. The current 60-day limit was established to provide sufficient margin in spent fuel pool temperature calculations as a result of decay heat loads in Region 2 from discharged fuel assemblies (Reference 39). Although the spent fuel pool cooling system and, thus, the associated restriction on heat load prevent structural integrity damage to the spent fuel pool, they are not assumed to function to mitigate the consequences of a design basis accident (DBA). The restriction on heat load is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The restriction on heat load is a non-significant risk contributor to core damage frequency and offsite doses. Since this does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4) and no NRC Final Policy Statement technical specification screening criteria apply, this requirement is relocated to the TRM. This is a Ginna TS Category (iii) change.

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iv. TS 5.4.4 and 5.4.5 - These were revised consistent with References 29 and 39 to provide the amount, kind, and source of material which is stored in the canisters. This is a Ginna TS 5.4.4 Category (v.b) and 5.4.5 were not added the new specifications for the reasons discussed in item 47(c) change.iii above. These are Ginna TS Category (v.c) and (iii), respectively.

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v. TS 5.4 - This was revised to include descriptions of the SFP drainage system and capacity. This information is currently contained in the bases for this section. Since NUREG-1431, Chapter 4 does not contain any bases, this information has been relocated to the specification. This is a Ginna TS Category (i) change.

48. Technical Specification 5.5

i. TS 5.5 - The description of the waste treatment systems design features was not added. No screening criteria apply for the description of these features. Specific waste treatment systems features are either covered in the Technical Specification LCO's or have been relocated to other licensee controlled documents and, therefore, do not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR Chapter 11). This is a Ginna TS Category (iii) change.

49. Technical Specification 6.1

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- i. TS 6.1.1 - The requirement was revised to include a statement that the ~~Plant Manager~~ plant manager shall approve each proposed test, experiment or modification to structures, systems or components that affect nuclear safety. This is a Ginna TS Category (iv.a) change.
- ii. TS 6.1 - A new requirement (Specification 5.1.2) was added which establishes the requirement for Shift Supervisor responsibility. This is a Ginna TS Category (iv.a) change.

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iii. ~~The plant manager title was revised to be more generic consistent with Reference 62. See also item 50.ii below. This is a Ginna TS Category (vi) change.~~

50. Technical Specification 6.2

- i. Cross references to existing regulatory requirements are redundant and generally not incorporated into NUREG-1431. This is a Ginna TS Category (ii) change.
- ii. Plant specific management position titles in the current Technical Specifications are replaced with generic titles consistent with Reference 62. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed TS 5.3, and compliance details relating to the plant specific management position titles are identified in licensee-controlled documents ~~the UFSAR~~. The two major specific replacements are the generic "Plant Manager" ~~plant manager~~ for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the a corporate executive responsible for overall plant nuclear safety" ~~vice president~~ in place of the specific Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant-controlled document ~~with the UFSAR which has specific regulatory review requirements for changes (e.g., as the UFSAR or QA Program)~~. This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change.
- iii. TS 6.2.1.d - The requirement describing the capability of training, health physics and quality assurance to have direct access to responsible corporate management ~~to support mitigation of their concerns was not added~~ modified to be consistent with NUREG-1431. Proposed TS 5.2.1 ~~These modifications are editorial changes only which do not change the intent or requirements of this specification. a requires that "lines of authority, responsibility and communication shall be established and defined throughout~~
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~~the highest management levels." The organizational structure is specified in the Ginna Station QA Program. Since changes to the QA Program are controlled by 10 CFR 50.54(a)(3), equivalent control is provided. This is a Ginna TS Category (ii) change. This is a Ginna TS Category (vi) change.~~

- iv. TS 6.2.2.b - The requirements describing the required operating crew compositions were not added. These requirements are specified in 10 CFR 50.54(k), (l), and (m) and proposed TS 5.2.2.a, 5.2.2.b, and 5.2.2.e. This is a Ginna TS Category (ii) change.
- v. TS 6.2.2.d - The requirement was revised to clarify that the individual qualified in radiation protection procedures is allowed to be absent for not more than two hours. This is consistent with the requirements for shift crew composition. This is a Ginna TS Category (v.c) change.
- vi. TS 6.2.2.e - The requirement describing the overtime requirement for plant staff who perform safety related functions was revised to reference a NRC approved program for controlling overtime. This is a Ginna TS Category (vi) change.

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 vii. A new requirement was added which specifies that the Operations Manager or Operations middle manager shall hold a SRO. This change is consistent with NUREG-1431 and ensures that at least one operations manager holds a SRO. This is a Ginna TS Category (iv.a) change.

51. Technical Specification 6.3

- i. TS 6.3.1 - The reference to the RG&E letter dated December 30, 1980, was replaced with wording considered more appropriate. The current STA program at Ginna Station is discussed in References 40 and 42 and was reviewed and approved by the NRC. The revised wording eliminates the need to revise the Technical Specifications if the STA program is later revised, but still requires NRC approval of these changes. This is a Ginna TS Category (vi) change.

52. Technical Specification 6.4 .

- i. TS 6.4 - The requirements for a Training Program were not added. The requirements are either adequately addressed by other Section 5.0 administrative controls or are addressed by 10 CFR 55 requirements. This is a Ginna TS Category (ii) change. Therefore, these requirements are relocated to the UFSAR. This is a Ginna TS Category (iii) change.

53. Technical Specification 6.5

None.

54. Technical Specification 6.6

None.

55. Technical Specification 6.7

- i. TS 6.7.1.a - The initial operator actions for Safety Limit (SL) violations were revised as follows:



- a. For violation of the Reactor Core or RCS Pressure SL in MODES 1 and 2, the requirement to immediately shutdown the reactor (effectively to be in MODE 3) was revised to allow 1 hour to restore compliance and place the unit in MODE 3. Immediately shutting down the reactor could infer action to immediately trip the reactor. The revision provides the necessary time to shutdown the unit in a more controlled and orderly manner than immediately tripping the reactor which could result in a plant transient. The proposed time continues to minimize the time allowed to operate in MODE 1 or 2 with a SL not met. This is a Ginna TS Category (v.b.44) change.
 - b. For violation of the RCS Pressure SL in MODES 3, 4, and 5, an additional action was added which requires restoring compliance with the SL within 5 minutes. Specifying a time limit for operators to restore compliance provides greater guidance to plant staff. This is a Ginna TS Category (v.a) change.
- ii. TS 6.7.1.b - The requirement for notification to management personnel and the offsite review function of a SL violation was not added to the new specifications. Notification requirements are relocated to the TRM. This is a Ginna TS Category (iii) change. The requirement for notification to the NRC of a SL violation was not added to the new specifications since this requirement is denoted in 10 CFR 50.36 and 10 CFR 50.72. This is a Ginna TS Category (ii) change.
- iii. TS 6.7.1.c - The requirement that a Safety Limit Violation Report be prepared was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the onsite review committee to review the Safety Limit Violation Report was not added to the new specifications. The responsibilities of the onsite review committee are relocated to the TRM. This is a Ginna TS Category (iii) change. SL violations are reported to the NRC in accordance with the provisions of 10 CFR 50.73. The details describing the requirements for content of the Safety Limit Violation Report is, therefore, controlled by the provisions of 10 CFR 50.73 and does not need to be specified in TS. This is a Ginna TS Category (ii) change.

- iv. TS 6.7.1.d - The requirement for the submittal of a Safety Limit Violation Report to the NRC was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR.50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the submittal of a Safety Limit Violation Report to management personnel and the offsite review function was not added to the new specifications. The distribution of reports submitted in accordance with 10 CFR 50.73 are relocated to the TRM. This is a Ginna TS Category (iii) change.

56. Technical Specification 6.8

- i. TS 6.8.1.d - The Offsite Dose Calculation Manual implementation is covered by a more generic item which is specified in Section 5.5. It is not necessary to specifically identify each program under procedures (see Section D, item 56.iv). Since the requirements remain, this is considered to be a change in the method of presentation only. This is a Ginna TS Category (i) change.
- ii. TS 6.8.1.e - The PCP description was not added since this program only implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71 and does not impose any new regulations. The detailed description of the PCP is provided in licensee controlled documents ~~with the requirement for the PCP relocated to the TRM.~~ This is a Ginna TS Category ~~(ii)~~ ⁽ⁱⁱⁱ⁾ change.
- iii. TS 6.8.1 - A new specification (TS 5.4.1.b) was added which establishes the requirement for written emergency operating procedures implementing the requirements of NUREG-0737 and NUREG-0737, Supplement 1. This is a Ginna TS Category (iv.a) change.

- iv. TS 6.8.1 - A new specification (TS 5.4.1.e) was added which establishes the requirement for written procedures for programs and manuals denoted in new Specification 5.5. These Programs include:

<u>ITS</u>	<u>Current TS</u>	<u>Program</u>
5.5.1	1.13 & 6.15	Offsite Dose Calculation Manual
5.5.2	4.4.3	Primary Coolant Sources Outside Containment
5.5.3	New	Post Accident Sampling Program
5.5.4	3.9 & 3.16	Radioactive Effluent Controls Program
5.5.5	New	Component Cyclic or Transient Limit
5.5.6	4.4.4	Pre-Stressed Concrete Containment Tendon Surveillance Program
5.5.7	4.2	Inservice Testing Program
5.5.8	4.2	Steam Generator (SG) Tube Surveillance Program
5.5.10	4.5:2.3 & 4.11.1	Ventilation Filter Testing Program
5.5.11	3.9.2.5 & 3.9.2.6	Explosive Gas and Storage Tank Radioactive Monitoring Program
5.5.12	4.6.1.d	Diesel Fuel Oil Testing Program
5.5.13	New	Technical Specification Bases Control
5.5.14	New	Safety Function Determination Program
5.5.15	New	Containment Leakage Rate Testing Program

The technical content of several requirements are being moved from other chapters of the current Technical Specifications and are proposed to be identified as Programs in accordance with the format of NUREG-1431. This is a Ginna TS Category (i) change. Other programs were added, except as discussed below, to ensure consistency in the implementation of required programs within the current licensing basis. The Radioactive Effluent Controls Program was added due to the relocation of the radiological Technical Specifications consistent with Generic Letter 89-01 and the changes to 10 CFR 20. The Bases Control program was added to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program was added to support implementation of the support system operability characteristics of the Technical Specifications (new LCO 3.0.6). These are Ginna TS Category (iv.a) changes.

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TS 6.8.1.c - The radiological environmental monitoring program is covered by a more generic item which is specified in specification 5.5. It is not necessary to specifically identify each program under procedures (see Section D, item 56.iv). Since the requirements remain, this is considered to be a change in the method of presentation only. This is a Gima TS category (3) change.

57. Technical Specification 6.9

- i. TS 6.9 - The reference to reporting requirements were revised consistent with 10 CFR 50.4. This is a Ginna TS Category (vi) change.
- ii. TS 6.9.1.1 - The requirement to submit a Startup Report was not added. The Startup Report is more appropriately addressed in the NRC Safety Evaluation Report authorizing an Operating License, increased power level, installation of a new nuclear fuel design or manufacturer, or modifications which significantly alter the nuclear, thermal, or hydraulic performances of the plant. The Startup Report is required to be submitted within 90 days following completion of the above activities and does not require NRC approval. Therefore, inclusion of the requirement for this report in Technical Specifications is not necessary to assure safe plant operation. This is a Ginna TS Category (ii) change.
- iii. TS 6.9.1.2 - The requirements describing the details of the monthly report were not added. These details are appropriately relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.
- iv. TS 6.9.1.3, TS 6.9.1.4, Table 6.9-1 and Table 6.9-2 - The details and methods implementing these specifications were not added. These details are appropriately relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. The submittal date was also changed to May 15th to allow the submittal of the Annual Radiological Environmental Operating Report to correspond with the Monthly Operating Report submittal date. This is a Ginna TS Category (iii) change.
- v. TS 6.9.1.4 - The specific date referenced for the annual submittal was revised consistent with the requirements of 10 CFR 50.36a. This is a Ginna TS Category (vi) change.
- vi. TS 6.9.1.5 - The requirement for the reporting of challenges to pressurizer PORVs or safety valves was revised from an annual to a monthly report and relocated to the Monthly Operating Report (new Specification 5.6.4). This is a Ginna TS Category (v.c) change.
- vii. TS 6.9.2.1 - The reporting requirement related to sealed sources was not added since this is specified in 10 CFR 30.50. ~~This is a Ginna TS Category (ii) change.~~ The detailed description of these reporting requirements are provided in licensee controlled documents. This is a Ginna TS Category (iii) change.

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viii. TS 6.9.2.4 - The reporting requirement for reactor overpressure protection system operation was revised. The reporting requirement is detailed in proposed Specification 5.6.4, and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. Since the criteria identified in 10 CFR 50.73 includes the area of degraded boundaries that necessitates reporting, any minor differences are negligible with regard to safety. This is a Ginna TS Category ~~(v.(ii) change.e) change~~.

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ix. A new requirement TS 5.6.5 was added which establishes the reporting requirement for the COLR. The COLR is required due to the removal of existing Technical Specification core operating limits. This is a Ginna TS Category (iv.a) change.

x. A new requirement TS 5.6.6 was added which establishes the reporting requirement for the RCS PTLR. The PTLR is required due to the removal of existing Technical Specification pressure and temperature operating limits. This is a Ginna TS Category (iv.a) change.

58. Technical Specification 6.10

None.

59. Technical Specification 6.11

None.

60. Technical Specification 6.12

None.

61. Technical Specification 6.13

i. TS 6.13.1 - Plant specific position titles in the current Ginna Station TS were replaced with generic titles ~~(i). The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (e.g., radiation protection technician). The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in the UFSAR or QA Program which has specific regulatory review requirements for changes.~~ This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change.

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62. Technical Specification 6.14

None.

63. Technical Specification 6.15

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- i. TS 6.15.1.b - The approval process for ODCM changes was revised to clarify that the effective changes be approved by the ~~Plant Manager~~ plant manager instead of the onsite review function. Since the onsite review function reports to the Plant Manager, this is a conservative change. This is a Ginna TS Category (v.e) ~~a~~ change.
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64. Technical Specification 6.16

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- i. TS 6.16 - The process for changes to the PCP was not added to the new specifications since this program only implements the requirements of 10 CFR Part 20, 10 CFR Part 61, and 10 CFR Part 71 and does not impose any new requirements. The detailed description of the PCP is provided in licensee controlled documents ~~and the requirement for the program is relocated to the TRM.~~ This is a Ginna TS Category ~~(ii)~~ (iii) change.

65. Technical Specification 6.17

- i. TS 6.17 - The requirements for major changes to radioactive waste treatment systems was not added. Changes to these systems are controlled by 10 CFR 50.59. NRC notification of significant changes to these systems is addressed by 10 CFR 50.59(b)(2). Therefore, this specification is relocated to the TRM. This is a Ginna TS Category (iii) change.

66. New Requirements (Ginna TS Category (iv.a) Changes)

- i. LCO 3.4.1 and the associated surveillance requirements were added for DNB limits. This new requirement places limits on pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure that the minimum DNBR will be met for all analyzed transients.
- ii. LCO 3.7.3 and the associated surveillances were added for the MFW pump discharge valves (MFPDVs), MFW regulating valves, and the associated bypass valves. This new requirement specifies an isolation time of 80 seconds for the MFPDVs and 10 seconds for the remaining valves and requires them to be OPERABLE above MODE 4 to provide isolation capability as assumed in the accident analyses.
- iii. LCO 3.7.4 and the associated surveillance were added for the atmospheric relief valves (ARVs). The LCO requires that the ARVs be OPERABLE when RCS average temperature is > 500°F in MODE 3 to provide cooldown capability following a SGTR event as assumed in the accident analyses. A Surveillance to verify that each ARV is capable of opening and closing once every 24 months was also added.

iv. A COLR was developed which contains the actual limits for LCOs associated with reactor physic parameters that may change with each refueling. To prevent the need to revise Technical Specifications for parameters which are calculated using NRC approved methodology, Generic Letter 88-16 (Ref. 56) allows these limits to be relocated from the technical specifications. A copy of the proposed Ginna Station COLR is provided in Attachment F. The following parameters were relocated to the COLR:

- a. SHUTDOWN MARGIN
- b. MODERATOR TEMPERATURE COEFFICIENT
- c. Shutdown Bank Insertion Limit
- d. Control Bank Insertion Limits
- e. Heat Flux Hot Channel Factor
- f. Nuclear Enthalpy Rise Hot Channel Factor
- g. AXIAL FLUX DIFFERENCE
- h. ~~Overtemperature ΔT and Overpower ΔT Trip Setpoints~~ Not used
- i. RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits
- j. ~~Accumulator Boron Concentration~~ Not used
- k. ~~RWST Boron Concentration~~ Not used
- l. ~~Spent Fuel Pool Boron Concentration~~ Not used
- m. Refueling Boron Concentration

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v. A RCS PTLR was developed which contains the actual limits for LCOs associated RCS pressure and temperature limits and LTOP. To prevent the need to revise Technical Specifications for parameters which are calculated using NRC approved methodology, NUREG-1431 allows these limits to be relocated from the technical specifications. A copy of the proposed Ginna Station PTLR is provided in Attachment G. The following parameters were relocated to the PTLR:

- a. RCS Pressure and Temperature Limits
- b. Low Temperature Overpressure Protection (LTOP) System Enable Temperature
- c. LTOP Setpoint

67. License

- i. The license was revised to relocate requirements associated with Secondary Water Chemistry Monitoring Program, Systems Integrity, and Iodine Monitoring to Appendix A of the license (i.e., TS). Changes to both the license and TS require NRC approval such that there is no reduction in commitment with respect to this change. This is a Ginna TS Category (i) change.
- ii. Minor editorial changes were made to provide consistency within the license. These are administrative changes only which do not change the intent of the license. These are Ginna TS Category (vi) changes.

- iii. The exemption to 10 CFR 50.48(c)(4) was removed from the license since this exemption expired in 1986 and is no longer required. This is a Ginna TS Category (vi) change.
- iv. The exemption to 10 CFR 50.46(a)(1) was removed from the license since this exemption is no longer required since the ECCS models for Ginna Station have since been revised. This is a Ginna TS Category (vi) change.

E. SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D are organized into 6 categories and subcategories as necessary. These categories of changes are evaluated with respect to 10 CFR 50.92(c) and shown to not involve a significant hazards consideration as described below.

E.1 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - ADMINISTRATIVE CHANGES

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (i), (ii), (v.c), or (vi) changes do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes involve either (1) the relocation of requirements within the Technical Specifications to support consolidation of similar requirements, (2) the reformatting, renumbering or rewording of the existing Technical Specifications to provide consistency with NUREG-1431, (3) the deletion of duplicate regulatory requirements, or (4) minor changes to the Technical Specifications such that the changes do not involve any technical issues. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. These changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.2 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - RELOCATED SPECIFICATIONS

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iii) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document (e.g., Technical Requirements Manual or UFSAR) which will continue to be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions in the Administrative Controls Section of the Technical Specifications. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, the majority of changes are consistent with the Westinghouse Standard Technical Specification, NUREG-1431, which has been approved by the NRC Staff. Therefore, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety. For those requirements proposed to be relocated which are retained within NUREG-1431, the relocated items are similar in nature to other relocated requirements or are not credited in the accident analyses for Ginna Station.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.3 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - MORE RESTRICTIVE CHANGES

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.a) and (v.a) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes provide more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes do impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change in Section D, each change in this category is by definition providing additional restrictions to enhance plant safety. The change maintains requirements within safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.4 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - LESS RESTRICTIVE CHANGES

LESS RESTRICTIVE CHANGE CATEGORY (iv.b.1)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.b.1) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Required Actions of the Diesel Generator (DG) Loss of Power (LOP) start instrumentation (current Table 3.5-1, Functional Units # 18 and #19) from an action to shutdown to an action to restore the channel to an OPERABLE status or enter the applicable conditions for an inoperable DG. The start instrumentation function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the OPERABLE start instrumentation channels from performing their intended function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that are no more restrictive than actions for the loss of one DG. The change maintains requirements within safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (iv.b.2)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.b.2) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the actions for an inoperable DG to: (1) eliminate the testing of the OPERABLE DG if, within 24 hours, it can be determined that the OPERABLE DG is not inoperable due to a common cause failure, and (2) eliminate the requirement to test the OPERABLE DG once every 24 hours until the second DG is restored to OPERABLE status (TS 3.7.2.2.b.1). The testing requirements for an OPERABLE DG are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not degrade the capability of the OPERABLE DG from performing its intended function since some DG failures can be conclusively determined not to apply to a second DG without requiring excessive testing. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that prevent unnecessary DG starts which can potentially adversely affect DG reliability. The change maintains DG OPERABILITY requirements within the safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.1)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.1) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Refueling Frequency which is used to define CHANNEL CALIBRATION and other testing intervals, from 18 months to 24 months (TS 1.12 and 4.4.6.2). The Frequency between CHANNEL CALIBRATIONS is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. TS required equipment is current maintained under a Reliability Centered Maintenance program such that their failures are tracked and trended. In addition, instrumentation setpoints and equipment history have been verified to be acceptable with respect to this change. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The equipment testing intervals are increased, but they still must be maintained OPERABLE consistent with their TS requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.2)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.

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2) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the applicability associated with the RCS Safety Limits (SL) in MODE 6 (current TS 2.2). Adequate margin exists such that it is not possible to pressurize the RCS greater than the SL pressure while in MODE 6. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these limits are not credited for mitigation of any accident in the omitted condition. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation.



change

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~~Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

No
changes

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~
Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

- No change*
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.

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The change maintains requirements within current safety analyses since it is not possible to pressurize the RCS greater than the SL pressure while in MODE 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.3)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.3) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement for the pressurizer water level lower limit of 12% (current TS 3.1.1.5.a). This requirement relates to a reactor trip function that was removed at Ginna Station as a result of IE Bulletin 79-06A (Ref. 45). Therefore, this change does not significantly increase the probability of a previously analyzed accident nor significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant since the trip function has already been removed. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the pressurizer low level trip function is no longer credited. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.4)

No change

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.4) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the applicability and deletes requirements associated with the overpressurization protection function of the pressurizer safety valves in MODES 5 and 6 (current TS 3.1.1.3.a and TS 3.1.1.3.b). The pressurizer safety valves do not perform a safety function in the omitted conditions. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these limits are not credited for mitigation of any accident in the omitted conditions. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

No change

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since these valves do not perform a safety function in MODES 5 and 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.5)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.5) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change deletes the requirements associated with SG temperature and pressure variables (current TS 3.1.1.2 and TS 3.1.2.2). The temperature and pressure variables are not specifically modeled in the safety analysis except through the variables of RCS pressure, temperature, and flow which are addressed in the heatup and cooldown rates in the PTLR. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these SG variables are not credited for mitigation of any accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since all necessary heatup and cooldown rates are addressed by the PTLR. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.6)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.6) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the period of time (from 6 hours to 72 hours) continued operation is allowed prior to confirming through the performance of an engineering evaluation, the structural integrity of the RCS after exceeding pressure or temperature limits (current TS 3.1.2.1.c.1). The requirement is associated with a function that is not an assumed initiator for any accidents previously evaluated since the exceeded limits are subsequently restored. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this function is not credited for mitigation of any accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The change maintains requirements within current safety analyses since the time that out-of-condition limits are restore is not changed. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.



Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.7)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.7) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change provides a Note allowing the plant to change MODES if either the containment sump monitor or both the containment atmospheric radioactivity monitors are inoperable (current TS 3.1.5.1). The RCS LEAKAGE detection system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the system to perform its required function since some form of LEAKAGE detection must always remain OPERABLE under these circumstances or a plant shutdown commenced. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

No
change

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~~3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.

No
change
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The change maintains requirements within current safety analyses since some form of RCS LEAKAGE detection must remain OPERABLE in MODES 1, 2, 3, and 4. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.8)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.8) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows an additional 4 hours to correct administrative and other similar discrepancies in the SG Tube Surveillance Program before commencing a reactor shutdown (current TS 3.1.5.2.2). Administrative discrepancies in the SG Tube Surveillance Program are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the SG tubes to perform their intended function since the limit on SG tube leakage remains. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that allow restoration of minor administrative discrepancies without affecting any safety analysis assumptions with respect to SG tube leakage. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.9)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.9) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 72 hours to restore accumulator boron concentration to within acceptable limits versus 1 hour (current TS 3.3.1.1.b and 3.3.1.3). The accumulator boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the accumulator to perform its required function under these circumstances since it will only allow additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

No
Change

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The accumulator boron concentration is not as critical feature as other accumulator parameters (e.g., water volume) such that additional time for restoration does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.10)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.10) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 72 hours to restore accumulator boron concentration to within acceptable limits versus 1 hour (current TS 3.3.1.1.a and 3.3.1.2). The RWST boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the RWST to perform its required function under these circumstances since it will only allow additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

^{No}
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~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~

~~THE PRECEDING TEXT WAS MOVED~~

The RWST boron concentration is not as critical feature as other RWST parameters (e.g., water volume) such that additional time for restoration does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.11)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.11) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change: (1) allows both SI pump flow paths to be isolated for up to 2 hours to perform pressure isolation valve testing, and (2) allows up to 4 hours, or until the RCS cold legs exceed 375°F, to place into service ECCS pumps declared inoperable due to LTOP considerations (current TS 3.3.1.1.c). The ECCS System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change allows required testing to be performed on the ECCS and reduces the potential for a transient to challenge the LTOP System. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change allows required testing to be performed on the ECCS, reduces the potential for a transient to challenge the LTOP Systems, and are consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.12)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.12) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change provides an AOT of 72 hours for two inoperable post-accident charcoal filter trains (current TS 3.3.2.2). The system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the system to perform its required function under these circumstances. This will allow an additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation.

No change

~~THE FOLLOWING TEXT WAS MOVED~~

~~Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.

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The change maintains requirements within current safety analyses since the CRFC units which supply the post-accident charcoal filter trains may be removed from service for up to 7 days prior to initiating a plant shutdown. In addition, the 100% redundant CS trains must remain OPERABLE in this condition. Therefore, this change does not involve a significant reduction in a margin of safety.



Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.13)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.13) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the CCW heat exchanger requirements to ~~only require allow 1 heat exchanger to be OPERABLE inoperable for up to 31 days versus 24 hours~~ (current TS 3.3.3.1). The CCW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the CCW system to perform its required function under these circumstances since the heat exchanger is a passive device similar to the CCW piping. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the CCW piping is also a passive device, which if it were to fail, would result in the loss of the entire CCW System which has been analyzed with acceptable results. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.14)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.14) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the AOT for two motor driven AFW pumps, from 24 hours to 72 hours, to be consistent with that for the turbine driven AFW pump (current TS 3.4.2.1.b). The AFW system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the AFW system to perform its required function under these circumstances since the turbine driven AFW pump is fully capable of supplying both SGs. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the turbine driven AFW pump is fully capable of supplying both SGs. In addition, for accident conditions in which AFW is not immediately required (i.e., not required for 10 minutes), the SAFW System is available. Therefore, this change does not involve a significant reduction in a margin of safety.

No
CHANGE

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.15)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v) Not used (see Reference 3.0).~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.15) do not involve a significant hazards consideration as discussed below:~~

~~16. — Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. 16)~~

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~~Not used (see Reference 3.0). —~~

~~The proposed changes increase the Surveillance Test Intervals (STIs) and AOTs for instrumentation supporting a number of TS Functions. LESS RESTRICTIVE CHANGE CATEGORY (v. — There are no actual related modifications to any of the affected systems. However, the changes are expected to reduce the test related plant scrams, reduce the test induced wear on the equipment, and reduce the number of forced outages related to test activities. 17)~~

~~Not used (see Reference 3.0). — Therefore, there is no significant increase in the probability of occurrence of a previously evaluated accident. Westinghouse topical reports WCAP 10271 P A (Ref. 48) and WCAP 14333 (Ref. 30) and associated supplements showed that the effects of these extensions of STIs and AOTs, which produced negligible impact, are bounded by previous analyses. Further, the NRC has reviewed the reports associated with WCAP 10271 P A and approved the conclusions on a generic basis. Therefore, the change does not significantly increase the consequences of a previously evaluated accident.~~

~~2. — Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The design and functional operation of the affected equipment are not changed by the proposed revisions. The proposed changes affect only the STIs and AOTs and will not impact the function of monitoring system variables over the anticipated ranges for normal operation, anticipated operational occurrences, or accident conditions. Further, the proposed changes do not introduce any new modes of plant operation, make any physical modifications, or alter any operational setpoints. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not~~

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created.

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The impact of reduced testing, other than as addressed above, is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Implementation of the proposed changes is expected to result in an overall improvement in safety due to:~~

- ~~i. Reduced testing which results in fewer inadvertent reactor trips, less frequent actuation of ESF components, and greater equipment availability.~~
- ~~ii. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation resulting from less frequent distraction to attend to testing.~~

~~Therefore, the proposed changes do not significantly reduce the margin of safety.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.16)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.16) do not involve a significant hazards consideration as discussed below:~~

- ~~1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the AOT from 1 hour to 6 hours to place an inoperable DG LOP instrumentation channel in the tripped condition (current Table 3.5-1, Functional Units #18 and #19). This Function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Since the action is to place the channel in the tripped condition, the function will continue to perform its safety function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

2. ~~Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

19 ~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.17)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.17) do not involve a significant hazards consideration as discussed below:~~

1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change provides an exception to allow bypassing of an inoperable DG LOP instrumentation channel and to delay entry into a Condition for the channel being tested (current Table 3.5-1, Functional Units #18 and #19). This Function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change is expected to reduce the test related plant scrams, reduce the test induced wear on the equipment, and reduce the number of forced outages related to test activities. Since trip capability is maintained, the Function will continued to perform its safety function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

2. ~~Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

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~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

LESS RESTRICTIVE CHANGE CATEGORY (v.b.18)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.18) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the required channels for Diesel Generator (DG) Loss of Power (LOP) start instrumentation (current Table 3.5-1, Functional Units # 18 and #19) from individually specifying the loss of voltage and degraded voltage channels to requiring two channels of undervoltage per 480 V safeguards bus. The start instrumentation function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the OPERABLE DG LOP instrumentation channels from performing their intended function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change only clarifies the actual design of the DG LOP instrumentation without affecting the safety function of the specified channels. The requirement for a loss of voltage and degraded voltage function is specified in the surveillance requirement for this LCO. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.19)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. Not used (see Reference 30)).~~

~~b. LESS RESTRICTIVE CHANGE CATEGORY (v.19) do not involve a significant hazards consideration as discussed below:~~

~~b.20)~~

~~Not used (see Reference 30)~~

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 21)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 21) do not involve a significant hazards consideration as discussed below:~~

~~1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the current AOT to restore inoperable Post Accident Monitors (PAMs), revises the actions for inoperable PAMs that are not restored to service within the AOT, and revises the PAM testing frequencies (current TS 3.5.3, 3.6.4.2, and 4.4.7). The PAMs are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The PAMS are not required to provide automatic response to any design basis accident. The additional time and surveillance frequencies has been evaluated and determined by the NRC to not significantly affect the contribution of the monitors to risk reduction. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

19. Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 22)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 22) do not involve a significant hazards consideration as discussed below:

120. 1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change ~~revises~~ allows the Required Actions for an inoperable reactor trip breaker use of a closed system to be used to allow 1 hour to restore the inoperable breaker before requiring a plant shutdown isolate a penetration with a failed containment isolation valve for up to 72 hours (current Table 3.5-1, Functional Unit #20) TS 3.6.3). The reactor trip breakers are only containment isolation system is not considered as an initiator for any accidents previously analyzed transients with respect to their spurious opening. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade, under the circumstances, the capability of the reactor trip breaker from performing containment isolation system to perform its intended required function under these circumstances since the closed system is a passive device which is missile protected. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new mode of plant operation or different kind of accident from any accident previously evaluated. The proposed change introduces no changes in the methods governing normal plant operation.

Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.

The change allows a short period to restore operation of Ginna Station in accordance with the inoperable reactor trip breaker before requiring a plant shutdown. The proposed change does not involve a significant reduction in a margin of safety. This time to restore the inoperable breaker. The containment isolation system remains capable of performing its intended function since the closed system is consistent with NUREG 1431. The missile protected, leak tested, and capable of maintaining containment integrity in the event of an accident. Therefore, this

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change does not involve a significant reduction in a margin of safety.

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~~This change is also consistent with NUREG 1431 which~~ Based upon the above information, it has been approved by determined that the NRC Staff proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety.

~~Based upon the above information~~ therefore, it has been determined is concluded that the proposed changes to meet the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, requirements of 10 CFR 50.92(c) and does do not involve a significant reduction in a margin of safety hazards consideration.—

Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration LESS RESTRICTIVE CHANGE CATEGORY (v.

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LESS RESTRICTIVE CHANGE CATEGORY (v.b. 23)

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Not used.

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 24)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 20)

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The proposed changes to the Ginna Station Technical Specifications 24) do not involve a significant hazards consideration as discussed in Section D and denoted by Category (v. below)

ii. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.~~ 20) do not involve a significant hazards consideration as discussed below:

The change revises the AOT for an inoperable 480 V safeguards bus from 1 hour to 8 hours before requiring a plant shutdown (current TS 3.7.2.2.—Operat

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increase the probability of a previously analyzed accident. The proposed change does not further degrade, under the circumstances, the capability of the Automatic Trip Logic (or reactor trip breaker) from performing its intended 480 V safeguards buses to perform their required

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2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) ~~or changes in the methods governing normal plant operation.~~ ~~The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation.~~ Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. ~~The change allows a period of~~ ~~allowing additional~~ time to restore the ~~an~~ inoperable Automatic Trip Logic and reactor trip breaker before requiring a plant shutdown. ~~480 V safeguards bus does not adversely affect the accident analyses since a redundant train is available.~~ The primary accident of concern during MODES 3, 4, and 5 is the rod ejection accident which is very unlikely due to the reduced system pressures and temperatures. ~~Increased time is also consistent with NUREG-1431.~~ Therefore, this change does not involve a significant reduction in a margin of safety.

~~Based upon the above information, it~~ ~~this change is also consistent with NUREG-1431 which has been determined that~~ ~~approved by the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety.~~ NRC Staff. —

Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.21)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.21) do not involve a significant hazards consideration as discussed below:~~

- ~~1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the current AOT to restore inoperable Post Accident Monitors (PAMs), revises the actions for inoperable PAMs that are not restored to service within the AOT, and revises the PAM testing frequencies (current IS 3.5.3, 3.6.4.2, and 4.4.7). The PAMs are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed~~

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~~change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 25)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 25) do not involve a significant hazards consideration as discussed below:

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1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to verify power distribution after each refueling from prior to reaching 50% RTP to < 75% RTP (current TS 3.10.2.1). Peaking factors are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. ~~The PAMs are not required to provide automatic response to any design basis accident. The additional time and surveillance frequencies has been evaluated and determined by the NRC to not significantly affect the contribution of the monitors to risk reduction. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Allowing power ascension to 75% RTP before verifying power distribution still provides the necessary margin to ensure design limits are met since peaking factors are most decreased near 100% RTP. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

LESS RESTRICTIVE CHANGE CATEGORY (v.b.22)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.22) do not involve a significant hazards consideration as discussed below:~~

1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows the use of a closed system to be used to isolate a penetration with a failed containment isolation valve (current TS 3.6.3). The containment isolation system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not~~

~~significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the containment isolation system to perform its required function under these circumstances since the closed system is a passive device which is missile protected. 26)~~

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.26) do not involve a significant hazards consideration as discussed below:

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- I. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to maintain F_{01} and F_{02} within limits at all times to only in MODE 1 (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. ~~The containment isolation system remains capable of performing its intended function since the closed system is missile protected, leak tested, and capable of maintaining containment integrity in the event of an accident. Therefore, this change does not involve a significant reduction in a margin of safety.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. These power distribution limits are not necessary to be met during MODE 2 since there is insufficient energy in the fuel to require these limits. In MODES 3, 4, 5, and 6, the reactor is not critical and, as such, these limits are not required. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

LESS RESTRICTIVE CHANGE CATEGORY (v.b.23)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.23) do not involve a significant hazards consideration as discussed below:~~

1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to have two offsite power sources OPERABLE prior to going above 350°F (current TS 3.7.2.1.b.2, 3.7.2.2.a, and 3.7.2.2.b). The current TS only require two sources in order to change MODES but allow indefinite operation once the MODE has been changed. Therefore, totally eliminating this requirement does not~~

~~significantly increase the probability of a previously analyzed accident nor significantly increase the consequences of a previously analyzed accident.~~

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The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.27) do not involve a significant hazards consideration as discussed below:

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1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Completion Time, from 24 hours to 72 hours, to reduce the Overpower ΔT , Overtemperature ΔT , and Power Range Neutron Flux - High trip setpoints when F_{α} or F_{β} is not within limits (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances since the Required Actions for these power distribution limits already require a power reduction in direct relationship to the percentage that the limit was exceeded. The reduction of trip setpoints only provides additional protection. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. ~~Only requiring one offsite power source does not adversely affect the accident analyses since offsite power is only assumed available if it results in worse consequences (e.g., steam line break). In addition, a second source of offsite power is available by backfeeding through the main transformer. Therefore, this change does not involve a significant reduction in a margin of safety.~~

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Allowing additional time to reduce the setpoints for associated reactor trip functions only provides secondary protection with respect to potential unanalyzed power distributions since reactor power has already been reduced. Therefore, this change does not involve a significant reduction in a margin of safety. The change for Overpower ΔT is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.24)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.24) do not involve a significant hazards consideration as discussed below:

1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the AOT for an inoperable 480 V safeguards bus from 1 hour to 8 hours before requiring a plant shutdown (current TS 3.7.2.2.c). The 480 V safeguards buses are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the 480 V safeguards buses to perform their required function under these circumstances since a redundant train is available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

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The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v-b-28) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change deletes the requirement to identify the cause of QPTR exceeding 1.02 or limit power to < 50% RTP (current TS 3-10.2.4). The QPTR is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. If the QPTR is not within limits, thermal power is required to be reduced proportional to the percentage that QPTR is outside the limits to compensate for the tilt and flux mapping must be initiated. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

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2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. ~~Allowing additional time to restore an inoperable 480 V safeguards bus does not adversely affect the accident analyses since a redundant train is available. The increased time is also consistent with NUREG 1431. Removing the requirement to identify the cause of the tilt or reduce power to < 50% RTP does not adversely affect the accident analyses since a power reduction proportional to the percentage that OPIR is outside the limit is required. It is not always possible to identify the cause of the tilt and the remaining Required Actions already underway are adequate to assure safe operation of the plant. This power change is consistent with NUREG-1431 and WCAP-12159 (Ref. 51). Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.~~

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

— LESS RESTRICTIVE CHANGE CATEGORY (v.b.25)

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.25) do not involve a significant hazards consideration as discussed below:~~

1. ~~29~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.29) do not involve a significant hazards consideration as discussed below:~~

~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to verify power distribution after each refueling from prior to reaching 50% RTP to < 75% RTP (current TS 3.10.2.1). Peaking factors are not considered as an initiator for any accidents previously analyzed. The change revises the Frequency for performance of control rod exercises from monthly to every 92 days (current Table 4.1-2, Functional Unit #6a). Control Rods are only considered as an initiator for rod ejection accidents which are not related to this Surveillance. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~

The proposed change does not further degrade the capability of the system

to perform its required function since this Surveillance only confirms normal operational indications of control rod OPERABILITY. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. ~~Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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- ~~3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~
 2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Allowing power ascension to 75% RTP before verifying power distribution still provides the necessary margin to ensure design limits are met since peaking factors are most decreased near 100% RTP. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.26)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.26) do not involve a significant hazards consideration as discussed below:~~

- (19)
- ~~1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to maintain F_Q and F_{SM} within limits at all times to only in MODE 1 (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~
 - ~~2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. These power distribution limits are not necessary to be met during MODE 2 since there is insufficient energy in the fuel to require these limits. In MODES 3, 4, 5, and 6, the reactor is not critical and, as such, these limits are not required. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.27)~~

① ~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.27) do not involve a significant hazards consideration as discussed below:~~

1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Completion Time, from 24 hours to 72 hours, to reduce the Overpower ΔT , Overtemperature ΔT , and Power Range Neutron Flux High trip setpoints when F_{α} or F_{β} is not within limits (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances since the Required Actions for these power distribution limits already require a power reduction in direct relationship to the percentage that the limit was exceeded. The reduction of trip setpoints only provides additional protection. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~
2. ~~Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Allowing additional time to reduce the setpoints for associated reactor trip functions only provides secondary protection with respect to potential unanalyzed power distributions since reactor power has already been reduced. Therefore, this change does not involve a significant reduction in a margin of safety. The change for Overpower ΔT is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.28)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.28) do not involve a significant hazards consideration as discussed below:~~

- (19)
1. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change deletes the requirement to identify the cause of QPTR exceeding 1.02 or limit power to < 50% RTP (current TS 3.10.2.4). The QPTR is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. If the QPTR is not within limits, thermal power is required to be reduced proportional to the percentage that QPTR is outside the limits to compensate for the tilt and flux mapping must be initiated. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.~~
 2. ~~Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

- ~~3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Removing the requirement to identify the cause of the tilt or reduce power to < 50% RTP does not adversely affect the accident analyses since a power reduction proportional to the percentage that QPTR is outside the limit is required. It is not always possible to identify the cause of the tilt and the remaining Required Actions already underway are adequate to assure safe operation of the plant. This power change is consistent with NUREG 1431 and WCAP 12159 (Ref. 51). Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG 1431 which has been approved by the NRC Staff.~~

~~Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.~~

~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.29)~~

~~The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.29) do not involve a significant hazards consideration as discussed below:~~

- 19
- ~~1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performance of control rod exercises from monthly to every 92 days (current Table 4.1 2, Functional Unit #6a). Control Rods are only considered as an initiator for rod ejection accidents which are not related to this Surveillance. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Control Rod OPERABILITY is normally verified by normal operational practices such that increasing the allowed Surveillance interval does not involve a significant reduction in a margin of safety. The change is also consistent with NUREG-1431 and NUREG-1366 (Ref. 8).~~

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.30)

The proposed changes to the Ginna Station Technical Specifications as

discussed in Section D and denoted by Category (v.b.30) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying the NaOH concentration in the spray additive tank from monthly to once every 184 days (current Table 4.1-2, Functional Unit #13). The spray additive tank is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function since the tank is passive with available level indications to the operators which would indicate a change in concentration. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the spray additive tank from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.31)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.31) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing an RCS water inventory balance from daily to once every 72 hours (current Table 4.1-2, Functional Unit #15). Verifying RCS water inventory is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of operations to identify LEAKAGE in the RCS since other indications, including letdown, are available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not degrade the capability of operations to identify LEAKAGE in the RCS since other indications are available. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.32)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.32) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing verification of the SFP boron concentration from once every 31 days to once every 31 days if a verification of fuel storage has not been complete (current Table 4.1-2, Functional Unit #17). Verifying SFP boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not affect the accident analyses since boron concentration is only credited during a fuel handling accident prior to the time which the fuel has been verified to be correctly stored. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). ~~The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not affect the assumptions used for a fuel handling accident. Therefore, this change does not involve a significant reduction in a margin of safety. This change (with the exception of the 31 day Frequency) is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.33)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.33) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying the DG fuel oil inventory from daily to once every 31 days (current Table 4.1-2, Functional Unit #16). The DG fuel oil tank is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function since the tank is passive with available level indications to the operators which would indicate a change in inventory. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. (no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the DG fuel oil tank from performing its intended safety function since other indicators are available to operators. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.~~

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

19 ~~LESS RESTRICTIVE CHANGE CATEGORY (v.b.34) LESS RESTRICTIVE CHANGE CATEGORY (v.b.34)~~

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.34) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying RCS gross specific activity from once every 72 hours to once every 7 days (current Table 4.1-4, Functional Unit #1). Verifying RCS gross specific activity is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of operations to identify fuel failures since other indications, including radiation alarms, are available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no ~~new or different type of equipment will be installed~~). The proposed ~~change introduces no new mode of plant operation or changes in the methods governing normal plant operation.~~ Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. (e.g., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not degrade the capability of operations to identify gross fuel failure since other indications are available. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.35)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.35) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to test the SI cold leg injection and RHR RCS PIVs each cold shutdown greater than 7 days (current TS 4.3.3.1). These valves are normally maintained closed (i.e., they are not relied upon or used during power operation or cold shutdown conditions). Performing testing on these PIVs should only be required once every 24 months or within 24 hours of their being opened since more frequent testing would not likely provide any additional information. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the PIVs to perform their required function since the valves are maintained closed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.



2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the RCS PIVS from performing their intended safety function since they will be tested a minimum of once every 24 months. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.36)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.36) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing the spray nozzle gas test from once every 5 years to once every 10 years (current TS 4.5.1.2.b). The spray ring nozzles are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the CS System to perform its required function since the nozzles are passive and located in a generally inaccessible area. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). ~~The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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~~3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

(19)

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the CS System from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. The revised Frequency is also consistent with NUREG-1431 and NUREG-1366 (Ref. 8).

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.37)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.37) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing actuation testing of the post-accident charcoal filter dampers from monthly to once every 24 months (current TS 4.5.2.3.5). The post-accident charcoal filters are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the post-accident charcoal filters to perform their required function since the dampers have demonstrated a high degree of reliability and the CS System provides a 100% redundant iodine removal capability. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

~~3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~

(19) 3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the post-accident filters from performing their intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.38)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.38) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing the spray additive valves from monthly once every 24 months (current TS 4.5.2.2.a). The spray additive valves are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the spray additive system from performing its required function since have demonstrated a high degree of reliability and the post-accident charcoal filters provide 100% redundant iodine removal capability. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the spray additive system from performing its intended safety function. The revised Frequency is also consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. The revised Frequency is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.39)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.39) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to test the AFW motor driven pump cross-over motor operated isolation valves (current TS 4.8.3). The AFW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the AFW System since the cross-over isolation valves are not credited in the accident analysis. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety.~~ The deletion of the AFW cross-over isolation valves testing requirements does not prevent the AFW System from performing its intended safety function since the valves are not credited in the accident analysis. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.40)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.40) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change eliminates the need to perform a verification that the AFW pumps can start within 10 minutes once every 18 months (current TS 4.8.10). The AFW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the AFW System from performing its required function since this verification is not consistent with the accident analysis times. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The deletion of this Surveillance does not prevent the AFW System from performing its intended safety function since the 10 minute verification is not consistent with the accident analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.41)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.41) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying a RHR pump is providing forced flow in MODE 6 from once every 4 hours to once every 12 hours (current TS 4.11.2.1). Verification of RHR pump OPERABILITY is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the RHR System to provide decay heat removal since there are numerous indications available to plant operators of a loss of an RHR pump. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the RHR System from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.42)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.42) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to perform Inservice Testing surveillances of the RHR pumps during MODES 5 and 6 (current TS 4.11.2.2). At least one RHR pump is operating and the breakers of the second pump are verified during these conditions such that performance of this test is only a duplication of existing surveillances. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the RHR System to provide decay heat removal since there are alternate Surveillances verifying pump OPERABILITY. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). ~~The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

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3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The deletion of this Surveillance does not prevent the RHR System from performing its intended safety function since the Inservice Testing Surveillance is mainly performed to verify pump operation at high pressures which do not exist in MODES 5 and 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.43)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.43) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change delays performance of the PORV functional channel test until 12 hours after decreasing to the LTOP enable temperature specified in the PTLR instead of within 31 days prior to entering this condition (current TS 4.16.1.a). The PORVs are only considered as an initiator for a previously analyzed accident with respect to spuriously opening. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change only provides a short period of time to verify that the PORV is OPERABLE for its LTOP functions since the PORV provides alternate functions, with different setpoints, in higher MODES. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The short period provided to perform the PORV testing ensures that the PORV remains capable of performing its multiple functions through all required MODES. This period of time is consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.44)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.44) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 1 hour to restore compliance for violations of the Reactor Core or RCS Pressure SL in MODES 1 and 2 instead of requiring an immediate shutdown of the plant (current TS 6.7.1.a). Since this change affects the Required Actions following a violation of SLs, this change does not significantly increase the probability of a previously analyzed accident. The proposed change only provides a short period of time to restore compliance before performing a shutdown of the plant in order to limit the potential for additional damage. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The short period provided to restore compliance provides operators with time to stabilize the plant before requiring a shutdown. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.45)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.45) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change increases the OPERABILITY tolerance for the pressurizer safeties from $\pm 1\%$ to $+ 2.4\%$, -3% (current TS 3.1.1.3.c.). Since the pressurizer safety valve setpoint remains above the normal operating pressure and the PORV setpoint, this change does not significantly increase the probability of a previously analyzed accident. The change has been evaluated with respect to the most limiting pressure transients and shown to be acceptable. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation since the pressurizer safety valve setpoints following testing remain $\pm 1\%$. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.



3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The increased OPERABILITY tolerance allows for setpoint drift which has been demonstrated to exist at Ginna Station. The increased tolerances have been analyzed for the most limiting pressure transients with safety limits still being met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.46)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.46) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change increases the fuel enrichment limit from 4.25 weight percent to 5.05 weight percent (current TS 5.3.1.b). The fuel enrichment limit is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change has been evaluated with respect to fuel handling accidents and shown to be acceptable with respect to offsite doses and 10 CFR 100. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.~~

(19)

3. ~~Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The increased fuel enrichment limit allows for Ginna Station to convert to 18 month cycles. The change has been analyzed and shown that all safety limits are still met. Therefore, this change does not involve a significant reduction in a margin of safety.~~

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.47)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.47) do not involve a significant hazards consideration as discussed below:

(115)

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows two SW pumps from the same electrical source to be inoperable for up to 72 hours (current TS 3.3.4.2). The inoperability of one SW train is not considered as an initiator for an accident previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Systems supported by SW and using the same electrical train as the two SW pumps are currently allowed 72 hours or more to restore one inoperable train. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change only provides consistency within the TS between the SW system and systems which it supports. Therefore, this change does not involve a significant reduction in a margin of safety.

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Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.48)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.48) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to "cease operations which may increase the reactivity of the core" (current TS 3.8.2) if the necessary containment penetrations are not isolated during refueling activities. Containment isolation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The required actions for not meeting the containment isolation provisions during refueling is to stop all CORE ALTERATIONS and fuel movement. This action precludes a fuel handling accident for which containment isolation prevents an offsite release. Requiring that all operations which may increase the reactivity of the core is not necessary since containment does not protect against this accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. No accident analyses are affected by the removal of this requirement. Therefore, this change does not involve a significant reduction in a margin of safety.

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Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.49)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.49) do not involve a significant hazards consideration as discussed below:

221
1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the testing requirements of radiation monitors R-11 and R-12 to only require a functional test of the purge valves on a refueling outage basis (every 24 months) versus quarterly (current TS Table 4.1-5, Functional Units #3a and #3b). The radiation monitors are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The radiation monitors actuate the Containment Ventilation Isolation System which is not credited in the accident analyses since it only serves to back up the containment isolation system. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The radiation monitors are not credited in any accident analysis. Therefore, this change does not involve a significant reduction in a margin of safety.

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Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.50)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.50) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to isolate containment if the RCS boron concentration is not maintained above 2000 ppm during refueling (current TS 3.6.1.b). Containment isolation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The required actions for not meeting the boron concentration limits is to stop all CORE ALTERATIONS and movement of irradiated fuel. This action precludes a fuel handling accident for which containment isolation prevents an offsite release. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

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2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. No accident analyses are affected by the removal of this requirement. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.51)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.51) do not involve a significant hazards consideration as discussed below:

218
1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises current requirements from restoring an inoperable manual AFW or SAFW pump initiation channel within 48 hours to declare the association pump train inoperable (current TS Table 3.5-2, Functional Unit #3.a and #3.f). The manual actuation of the AFW and SAFW pumps is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Since the manual initiation functions only affect one AFW or SAFW pump, entering the LCO for the affected pump is consistent with all other pump operability requirements. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change provides consistency within the TS without allowing a pump to be inoperable for a period greater than is currently allowed. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 52)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 52) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to reduce the high neutron flux trip setpoint to $\leq 85\%$ RTP when power is reduced to $\leq 75\%$ RTP with a misaligned rod (current TS 3.10.4.3.2.b and 3.10.4.3.2.c). The accident analyses are performed assuming that one rod remains fully withdrawn following operation at full power conditions. Since reducing power to $\leq 75\%$ RTP and verifying peaking limits are still maintained must also be performed, the accident analyses remain valid. Therefore, this change does not significantly increase the probability of a previously analyzed accident and does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analyses assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v. b. 53)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v. b. 53) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change delays the determination of E until 31 days after a minimum of 2 EFPDs and 20 days of MODE 1 operation following the reactor being subcritical for ≥ 48 hours (current TS Table 4.1-4, Functional Unit #3). The determination of E is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The allowance of an additional 31 days ensures that a true representative sample is obtained such that the potential for false readings is reduced. The actual value of E is not changed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

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2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analysis assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.54)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.54) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows one diesel generator to be inoperable with no offsite power available for up to 12 hours (current TS 3.7.2-2.d). Since the loss of all offsite power and the failure of a diesel generator are assumptions of most accident analysis, this change does not significantly increase the consequences of an accident. The probability of an accident previously analyzed is not increased since offsite power and the diesel generators only mitigate an accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analysis assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

F. ENVIRONMENTAL CONSIDERATION

RG&E has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration as documented in Section D above;

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite since all specifications related to offsite releases are retained, addressed by existing regulations, or relocated to a licensee controlled program subject to the current regulations; and
3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since no new or different type of equipment are required to be installed as a result of this LAR, and the frequency of required testing which may result in radiation exposure is to be optimized consistent with industry practices.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

G. REFERENCES

1. NUREG-1431, *Standard Technical Specifications, Westinghouse Plants*, September 1993.
2. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Conversion to Improved Technical Specifications*, dated February 28, 1994.
3. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 37 to Full-Term Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. 52404)*, dated May 30, 1989.
4. Letter from R.A. Purple, NRC, to E.J. Nelson, RG&E, Subject: *Issuance of Amendment No. 5 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated February 13, 1975.
5. Letter from R.A. Purple, NRC, to E.J. Nelson, RG&E, Subject: *Issuance of Order for Modification of Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated December 27, 1974.
6. NUMARC 93-03, *Writer's Guide for the Restructured Technical Specifications*, February 1993.
7. WCAP-13749, *Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement*, May 1993.
8. Generic Letter 93-05, *Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation*, September 27, 1993.
9. Atomic Industrial Forum (AIF) General Design Criteria (GDC), Issued for comment July 10, 1967.
10. Letter from R.L. Laufer, NRC; to C.A. Schrock, WPS; Subject: *Amendment No. 110 to Facility Operating License No. DPR-43 (TAC No. M88374)*, dated August 3, 1994.
11. WCAP-14040, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, Revision 1, December 1994.
12. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *SEP Topic XV-9, Startup of an Inactive Loop, R.E. Ginna*, dated August 26, 1981.
13. NUREG-0452, *Westinghouse Standard Technical Specifications*.
14. Letter from D.L. Ziemann, NRC, to L.D. White, RG&E, Subject: *Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated July 26, 1979.
15. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *SEP Topics V-10.B, V-11.B, and VII-3*, dated September 29, 1981.

16. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves*, dated April 20, 1981.
17. Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Detection Systems*.
18. NUREG-0821, *Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant*, December 1982.
19. Generic Letter 84-04, *Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops*, February 1, 1984.
20. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Natural Circulation Cooldown, Generic Letter 81-21, R.E. Ginna Nuclear Power Plant*, dated November 22, 1983.
21. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 57 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated December 7, 1993.
22. Federal Register, Volume 60, page 9634, February 21, 1995.
23. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 54 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M77849)*, dated August 30, 1993.
24. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Containment Isolation Boundaries (TAC M77849)*, dated December 21, 1994.
25. Letter from G.E. Lear, NRC, to R.W. Kober, RG&E, Subject: *Containment Purge Technical Specifications, Issuance of Amendment No. 13 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated February 15, 1986.
26. NRC Temporary Inspection 2515/126, *Evaluation of On-Line Maintenance*.
27. Letter from D.M. Crutchfield, NRC, to L.D. White, RG&E, Subject: *Lessons Learned Category "A" Evaluation*, dated February 15, 1986.
28. Letter from J.A. Zwolinski, NRC, to R.W. Kober, RG&E, Subject: *TMI Action Plan Technical Specifications*, dated April 20, 1981.
29. Westinghouse, *Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters*, dated June, 1994.
30. ~~WCAP 14333, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times, May 1995~~ letter from R.

C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: Change to Technical Specification Instrumentation Requirements. Conversion to Improved Technical

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Specifications, dated August 31, 1995.

31. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: ~~Issuance of Amendment—NoChange to Technical Specification Instrumentation Requirements, Conversion to Improved Technical Specifications, dated August 31, 1995.~~

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~~47 to Facility Operating License No. DPR-18—R.E. Ginna Nuclear Power Plant (TAG No. 77515), dated November 19, 1991.~~

32. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Diesel Generator Surveillance and Testing*, dated April 23, 1981.
33. NUREG-0944, *Safety Evaluation Report Related to the Full-Term Operating License for R.E. Ginna Nuclear Power Plant*, dated October 1983.
34. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Safety Evaluation for Ginna - SEP Topic VIII-2*, dated June 24, 1981.
35. Letter A.R. Johnson (NRC), to R.C. Mecredy (RG&E), Subject: *Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3*, dated February 24, 1993.
36. Letter from M.B. Fairtile, NRC, to R.W. Kober, RG&E, Subject: *Technical Specifications on Battery Discharge Testing*, dated May 8, 1986.
37. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Safety Evaluation for Ginna - SEP Topic VIII-3A*, dated July 31, 1981.
38. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 51 to Operating License No. DPR-18*, dated April 13, 1993.
39. Letter from J.A. Zwolinski, NRC, to R.W. Kober, RG&E, Subject: *Increase of the Spent Fuel Pool Storage Capacity*, dated November 14, 1984.
40. Letter from W.A. Paulson, NRC, to R.W. Kober, RG&E, Subject: *Plant Staff Working Hours and Reporting Requirements for Safety Valve and Relief Valve Failures and Challenges*, dated January 31, 1984.
41. Letter from R.W. Kober, RG&E, to M. Fairtile, NRC, Subject: *Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04)*, dated May 14, 1986.
42. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *NUREG-0737, Item I.A.1.1, Shift Technical Advisor*, dated October 12, 1989.
43. Letter from D.M. Crutchfield, NRC, to L.E. White, RG&E, Subject: *Issuance of Amendment No. 33 to Provisional Operating License No. DPR-18*, dated June 13, 1980.
44. Letter D.M. Crutchfield (NRC) to J.E. Maier (RG&E), *Safety Evaluation for Ginna: SEP Topic VII-6*, dated June 24, 1981.
45. IE Bulletin 79-06A, *Review of Operational Errors and System Misalignments Identified During TMI Incident*.
46. Letter from D.L. Ziemann, NRC, to L.D. White, RG&E, Subject: *Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated June 15, 1979.

47. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Generic Letter 88-20*, dated March 15, 1994.
48. WCAP-10271-P-A, Supplement 2, Rev.1, June 1990.
49. Letter from D.M. Crutchfield (NRC) to J. Maier (RG&E), Subject: *Fuel Handling Accident Inside Containment*, dated October 7, 1981.
50. WCAP-13029, *MERITS Program, Phase III, Comments on Draft NUREG-1431, Standard Technical Specifications Westinghouse Plants*, July 1991.
51. WCAP-12159, *MERITS Program, Phase II, Technical Specifications and Bases*, March 1989.
52. WCAP-11618, *MERITS Program, Phase II, Task 5, Criteria Application*, November 1987.
53. ASME, Boiler and Pressure Vessel Code, Section XI.
54. EG&E Report, EGG-NTAP-6175, *In-Service Leak Testing of Primary Pressure Isolation Valves*, February 1983.
55. Letter from V.L. Rooney, NRC, to J.F. Opeka, Northeast Nuclear Energy Company, Subject: *Issuance of Amendment No. 105 (TAC No. M89518)*, dated February 22, 1995.
56. Generic Letter 88-16, *Removal of Cycle-Specific Parameter Limits from Technical Specifications*, dated October 4, 1988.
57. Letter from A.G. Hansen, NRC, to R.E. Link, Subject: *Amendment Nos. 157 and 161 to Facility Operating License Nos. DPR-24 and DPR-27 (TACS M85689 and M85690)*, dated December 8, 1994.
58. Ginna Station LER 95-001, Subject: *Pressurizer Safety Valve Lift Settings Found Above Technical Specification Tolerance During Post-Service Test Due to Setpoint Shifts, Results in Independent Train Being Considered Inoperable*, dated March 6, 1995.
59. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3 (TAC No. M80439)*, dated February 24, 1993.
60. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Generic Letter 90-06, Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors,"* dated September 15, 1992.
61. Letter from R.E. Smith, RG&E, to C. Stahle, NRC, Subject: *Change P-10 Permissive*, dated December 22, 1988.

62. Letter from C.I. Grimes, NRC, to NSSS Owners Groups, Subject: *Use of*



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Generic Titles in STS," dated November 10, 1994.

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Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: Application for Amendment to Facility Operating License, Implementation of 10 CFR 50, Appendix J, Option B, dated November 27, 1995.

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Letter from M. Modes, NRC, to R.C. Mecredy, RG&E, Subject: 10 CFR 50.54 Quality Assurance Program Change Review," dated March 22, 1995.

Attachment M

Location of Relocated Current Technical Specification Requirements

ATTACHMENT J

Documentation of Changes to May 26, 1995 Submittal

ITS RELATED PROBLEMS POST 5/26/95 SUBMITTAL

20-Dec-95

ITEM #: 1

CHAPTER/LCO: 3.5.4

DESCRIPTION OF ISSUE: Must add an LCO 3.0.6 exemption since the SFDP would imply that the ECCS would have to be declared inoperable, thus entering LCO 3.0.3, if the RWST were inoperable.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 2/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: See #129.

ITEM #: 2

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Must add an LCO 3.0.6 exemption since the SFDP would imply that the CS System would have to be declared inoperable, thus entering LCO 3.0.3, if the spray additive tanks were inoperable.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: See #129.

ITEM #: 3

CHAPTER/LCO: 3.7.6

DESCRIPTION OF ISSUE: Must add an LCO 3.0.6 exemption since the SFDP would imply that the AFW System would have to be declared inoperable, thus entering LCO 3.0.3, if the CSTs were inoperable.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: See #129.

ITEM #: 4

CHAPTER/LCO: 3.8.6

DESCRIPTION OF ISSUE: RG&E attempted to relocate the battery parameter table from the LCO since we currently do not have this requirement. However, the LCO directly references this table such that NRC approval would be required prior to make changes to these parameters. Revise LCO to correct this problem.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 4/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Deleted table in its entirety and revised LCO based on 11/15/95 meeting.

ITEM #: 5

CHAPTER/LCO: 3.9.5

DESCRIPTION OF ISSUE: Must add a note to the LCO to allow testing of single drop line valves for up to 12 hours since any MOV work while in MODE 6 would require declaring both RHR pumps inoperable. Since these are PIVs, this testing can only be done < 200F which includes MODE 6.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: LCO already allows 1 hour for this scenario. Submit LAR at future date if needed.

ITEM #: 6

CHAPTER/LCO: 3.3.3

DESCRIPTION OF ISSUE: The bases for LCO 3.7.6 state that only 1 of 2 CSTs is required to meet this LCO. There is only one train of level indication per CST. However, LCO 3.3.3 requires two trains of CST to be available. Revise LCO 3.3.3 to add a Note to clarify this requirement.

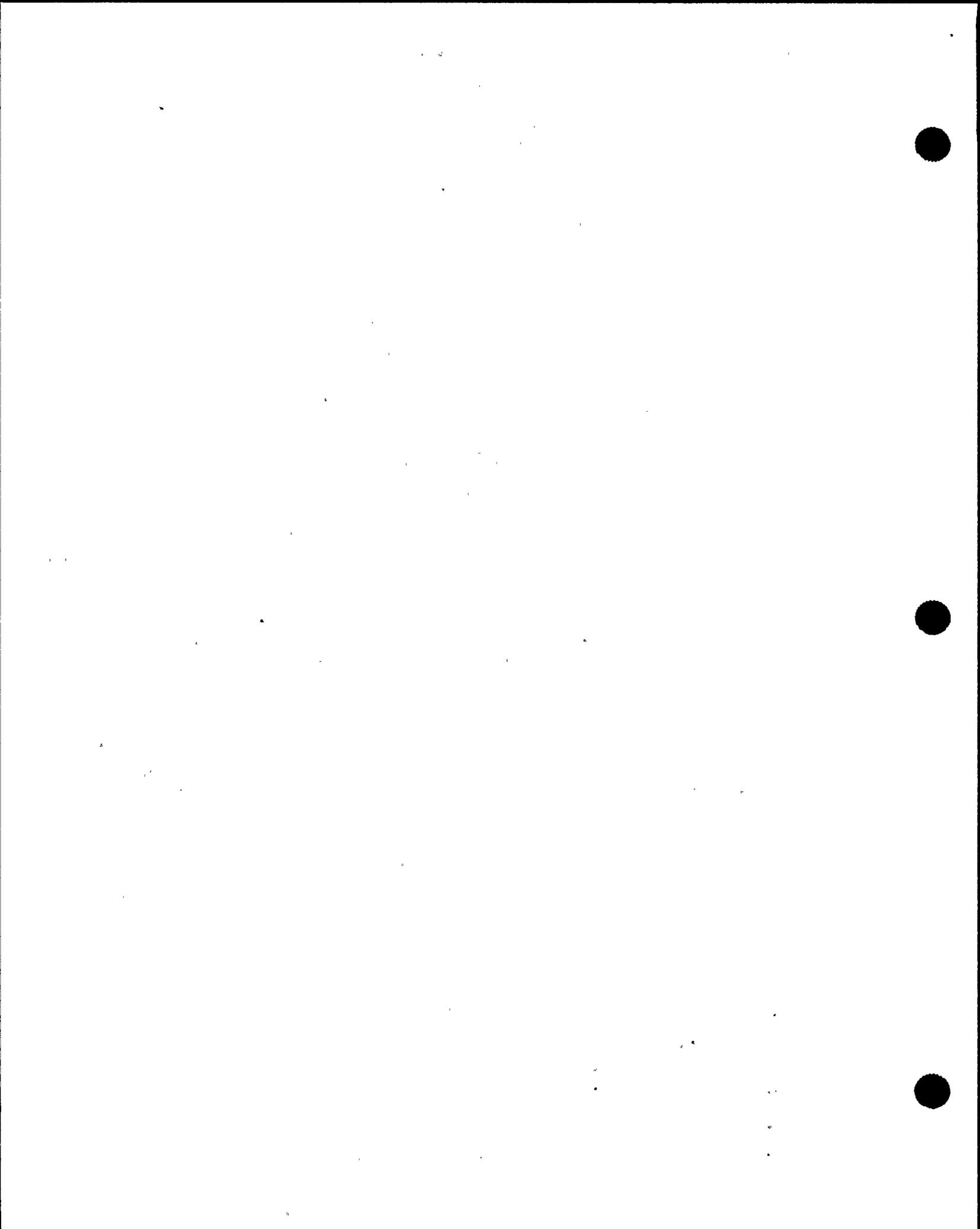
DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 9/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Vogtle currently has similar note.



20-Dec-95

ITEM #: 7

CHAPTER/LCO: 3.3.4

DESCRIPTION OF ISSUE: Revise SR 3.3.4.2 to relocate the undervoltage and degraded voltage DG LOP Instrumentation setpoints.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Traveller never generated to support this change.

ITEM #: 8

CHAPTER/LCO: 5.2.2.d

DESCRIPTION OF ISSUE: Remove the overtime restrictions for licensed personnel.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: No basis for change exists. Therefore, left ITS as is.

20-Dec-95

ITEM #: 9

CHAPTER/LCO: 3.7.7

DESCRIPTION OF ISSUE: Add a Note to Condition C, similar to the note for Condition G of LCO 3.7.5, to prevent the SFDP from requiring entry into LCO 3.0.3 with the loss of CCW.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 10/23/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Tracked by #100

ITEM #: 10

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Add a Note to Condition C, similar to the note for Condition G of LCO 3.7.5, to prevent the SFDP from requiring entry into LCO 3.0.3 upon loss of SW.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 10/23/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Tracked by #102

ITEM #: 11

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Peach Bottom was required to submit a letter acknowledging that several commitments and SERs were based on the fact that items were in the TS which are now being relocated. The letter states that these items are now adequately controlled by 50.59 and 50.54.

DATE IDENTIFIED: 6/30/95

DATE CLOSED:

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Required prior to implementation.

ITEM #: 12

CHAPTER/LCO: 3.8.6

DESCRIPTION OF ISSUE: SR 3.8.6.3 requires verification every 92 days that the average electrolyte temperature of representative battery cells is $\geq 65F$. This is a new SR for Ginna Station. Electrical Engineering has since concluded that $\geq 55F$ is acceptable. This temperature is in []s within the NUREG.

DATE IDENTIFIED: 6/30/95

DATE CLOSED: 12/ 4/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 13

CHAPTER/LCO: 3.3.1

DESCRIPTION OF ISSUE: There are several format and technical "brokes" within this LCO.

DATE IDENTIFIED: 7/13/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #222.

ITEM #: 14

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: There are several format and technical "brokes" within this LCO. These were attempted to be fixed in the 5/26/95 submittal.

DATE IDENTIFIED: 7/13/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #218.

20-Dec-95

ITEM #: 15

CHAPTER/LCO: 3.7.11

DESCRIPTION OF ISSUE: The current LCO has an exception to LCO 3.0.3 and is missing one for LCO 3.0.4.

DATE IDENTIFIED: 7/13/95

DATE CLOSED: 10/13/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Addressed by #148.

ITEM #: 16

CHAPTER/LCO: 3.7.12

DESCRIPTION OF ISSUE: The current LCO has an exception to LCO 3.0.3 and is missing one for LCO 3.0.4.

DATE IDENTIFIED: 7/13/95

DATE CLOSED: 10/13/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Addressed by #148.

20-Dec-95

ITEM #: 17

CHAPTER/LCO: 3.7.13

DESCRIPTION OF ISSUE: The current LCO has an exception to LCO 3.0.3 and is missing one for LCO 3.0.4.

DATE IDENTIFIED: 7/13/95

DATE CLOSED: 10/13/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Addressed by #148.

ITEM #: 18

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment E of submittal. This closes comments: 3.3Q3, 3.3Q10, 3.3Q11, 3.3Q12, 3.3Q13, 3.3Q15, 3.3Q32, 3.3Q36, 3.3Q40, 3.3Q43, 3.3Q45, 3.3Q47, and 3.3Q48

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Corrections were made to Attachment A as necessary such that no further changes to Attachment E are required.

ITEM #: 19

CHAPTER/LCO: 3.3.1

DESCRIPTION OF ISSUE: Remove all changes justified by WCAP-14333 and implement TOPS (WCAP-10271). This closes comments 3.3Q7, 3.3Q16, 3.3Q17, 3.3Q19, 3.3Q20, 3.3Q21, 3.3Q22, 3.3Q27, 3.3Q34, 3.3Q23, 3.3Q41, 3.3Q42, and 3.3Q49.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #222.

ITEM #: 20

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment B of submittal. This closes comments: 3.3Q12, 3.3Q14, 3.3Q21, 3.3Q25, 3.3Q26, 3.3Q31, 3.3Q40, 3.3Q45, 3.3Q46, 3.3Q47, and 3.3Q50.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



ITEM #: 21

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Remove all changes justified by WCAP-14333 and implement TOPS (WCAP-10271). This closes comments: 3.3Q33, 3.3Q35, 3.3Q37, 3.3Q41, 3.3Q42, 3.3Q52, and 3.3Q53.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed per #218.

ITEM #: 22

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Revise the description of the AFW Functions (#6) to clarify which pumps are actuated by each signal. This closes comment 3.3Q28.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #218.

20-Dec-95

ITEM #: 23

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Revise the bases justification for the 48 hour Completion Time for Required Action B.1 with respect to the Trip of the MFW Pumps for actuating the motor-driven AFW pumps. This closes comment 3.3Q29.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 24

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment C of the submittal. This closes comments: 3.3Q30 and 3.3Q56.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 25

CHAPTER/LCO: 3.3.3

DESCRIPTION OF ISSUE: Revise the Required Channel column of Table 3.3.3-1 for the SG Water Level Narrow Range function to read "2 per SG." This provides consistency with the wording for the SG Wide Range requirement. This closes comment 3.3Q54.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 26

CHAPTER/LCO: 3.3.4

DESCRIPTION OF ISSUE: Add a drawing of the LOP DG Start Instrumentation to the bases to provide additional clarification of what is meant by channels. This closes comment 3.3Q55.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 27

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment A of submittal. This closes comment 3.3Q56.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This change was not made since all necessary changes to Attachment B were made.

ITEM #: 28

CHAPTER/LCO: 3.3.1

DESCRIPTION OF ISSUE: Request that Westinghouse provide a justification as to why use of only a Trip Setpoint column is acceptable versus the table Reviewer's Note. This closes comment 3.3Q2.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: On hold pending numerous instrumentation issues.

ITEM #: 29

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Revised Setpoint Analysis DA-EE-92-087-21 to use an Allowable Value of 1715 psig vs. 1711 psig for Safety Injection - Pressurizer Pressure - Low function. This closes comment 3.3Q38.

DATE IDENTIFIED: 7/27/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Revision 1 issued on 11/22/95.

ITEM #: 30

CHAPTER/LCO: 3.8.3

DESCRIPTION OF ISSUE: Revise Required Action A.1 to only allow 12 (or 8 hours) to restore the onsite DG fuel oil supply versus the current 48 hours. This closes comment 3.8Q13.

DATE IDENTIFIED: 8/11/95

DATE CLOSED: 12/ 4/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 31

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Revise the bases for AFW actuation on low SG level to include discussion of the effect of the loss of Instrument Bus D.

DATE IDENTIFIED: 8/11/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 32

CHAPTER/LCO: 3.8

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment C of the submittal. This includes Conditon C of LCO 3.8.1 (CT column).

DATE IDENTIFIED: 8/12/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 33

CHAPTER/LCO: 3.8

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment A of the submittal. This closes comment 3.8Q22.

DATE IDENTIFIED: 8/24/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary changes to Attachment B were made such that no further changes to Attachment A were required.

ITEM #: 34

CHAPTER/LCO: 3.6

DESCRIPTION OF ISSUE: Either the NRC or the WOG must issue a traveller package for relocating notes from the Frequency column to the Surveillance column. The NRC has agreed that this is a "good practice" per telecon on 8/23/95. This closes comments 3.6Q1 and 3.6Q15.

DATE IDENTIFIED: 8/30/95

DATE CLOSED: 10/10/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: WOG rejected traveller as being below threshold. NRC likes idea but also not willing to generate a traveller. No further action required by RG&E.

20-Dec-95

ITEM #: 35

CHAPTER/LCO: 3.6

DESCRIPTION OF ISSUE: Revise the Containment Section LCOs and Bases in support of the new Appendix J rule based changes. Make sure all programmatic changes that are required are submitted and that Pa is correct (this affects 3.6Q10 and 3.6Q20). This closes comments 3.6Q4 and 3.6Q6.

DATE IDENTIFIED: 8/30/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 36

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Remove the text added to Required Action Note 2 for LCO 3.6.2, Conditions A and B (change C.58.ii). This closes comment 3.6Q14.

DATE IDENTIFIED: 8/30/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 37

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Add back into the bases for Conditions C.1, C.2, and C.3 the phrase "(e.g., only one seal per door failed)" which was removed by change C.58.v.b. This closes comment 3.6Q27.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 38

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Add the last sentence back to the first paragraph of the bases for SR 3.6.2.1 which was deleted by change C.58.v.b. This closes comment 3.6Q29.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: A modified sentence was added as a result of incorporative of #35.

ITEM #: 39

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Remove change C.58.x to Conditions C.2 and C.3 (made text plural).
This closes comment 3.6Q31.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 40

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Revise Actions Note #1 to include the phrase "except for Shutdown
Purge System valve flow paths..." This closes comment 3.6Q35.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 41

CHAPTER/LCO: 3.6

DESCRIPTION OF ISSUE: Correct typographical errors within the bases for Attachment D that are correctly shown in Attachment C. This closes comments 3.6Q74, 3.6Q77, and 3.6Q82.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary corrections to Attachment C were made such that no changes to Attachment D were required.

ITEM #: 42

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Relocate Action Table Note to Condition F with necessary Required Actions and Completion Times. This closes comment 3.6Q78.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 43

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Must add a new Surveillance similar to NUREG-1431 SR 3.6.7.1. This closes comment 3.6Q85.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 44

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Revise SR 3.6.6.13 to include the phrase "that is not locked, sealed, or otherwise secured in position." This closes comment 3.6Q86.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 45

CHAPTER/LCO: 3.6.7

DESCRIPTION OF ISSUE: Revise the hydrogen recombiner testing to include a physical and visual inspection of the units and verification that piping is not plugged, that the ignitor is OPERABLE, and the units are not fouled every 24 months. This will be a "functional check" versus a "functional test." This closes comments 3.6Q98 and 3.6Q99.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 46

CHAPTER/LCO: 3.6.7

DESCRIPTION OF ISSUE: Clarify the Background bases for LCO 3.6.7 with respect to operation of the hydrogen recombiners. This closes comment 3.6Q103.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 47

CHAPTER/LCO: 3.6.4

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment A of the submittal. This closes comments 3.6Q122 and 3.6Q140.

DATE IDENTIFIED: 8/31/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 48

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Expand submittal discussion to provide additional details on why LCOs and important bases discussions were not added.

DATE IDENTIFIED: 9/ 5/95

DATE CLOSED: 12/20/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: NRC agreed to withdraw request on 12/14/95.

20-Dec-95

ITEM #: 49

CHAPTER/LCO: 3.4

DESCRIPTION OF ISSUE: Correct typographical errors and perform minor editorial changes and clarifications within Attachment A of submittal. This closes comments 3.4Q1, 3.4Q4, 3.4Q5, and 3.4Q11.

DATE IDENTIFIED: 9/ 9/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 50

CHAPTER/LCO: 3.1

DESCRIPTION OF ISSUE: Correct typographical errors and perform minor editorial changes and clarifications to Attachment A of submittal. This closes comments 3.1Q3, 3.1Q7, and 3.1Q8.

DATE IDENTIFIED: 9/ 9/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 51

CHAPTER/LCO: 3.1.4

DESCRIPTION OF ISSUE: Revise Attachment A, Section D, item 20.xvi to be a "less restrictive change" with the associated no significant hazards evaluation. This closes comment 3.1Q4.

DATE IDENTIFIED: 9/9/95

DATE CLOSED: 12/8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 52

CHAPTER/LCO: 3.4.16

DESCRIPTION OF ISSUE: Revise Attachment A, Section D, item 28.iv.c to be a "less restrictive change" with the associated no significant hazards evaluation. This closes comment 3.7Q16.

DATE IDENTIFIED: 9/9/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 53

CHAPTER/LCO: 3.2.1

DESCRIPTION OF ISSUE: Track WOG Traveller concerning increased Completion Time for revising the Power Range Neutron Flux High trip function in both LCO 3.2.1 and 3.2.2. This closes comment 3.2Q2.

DATE IDENTIFIED: 9/10/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME: WOG-22

COMMENTS: Traveller under industry review.

ITEM #: 54

CHAPTER/LCO: 3.1.1

DESCRIPTION OF ISSUE: Expand bases for ASA in blocked text on page B3.1-3 to state "shutdown (MODE 5)." Also add discussion from CTS 3.10-11 concerning CTS Figure 3.10-2 to Insert 3.1.1.A.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 55

CHAPTER/LCO: 3.1.3

DESCRIPTION OF ISSUE: Separate SR 3.1.3.2 into 2 different SRs since you can fail different portions of the current SR and be in different conditions.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 56

CHAPTER/LCO: 3.1.3

DESCRIPTION OF ISSUE: Evaluate need for Note to Condition A and delete if no longer required.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Deleted note since determined not to be required.

20-Dec-95

ITEM #: 57

CHAPTER/LCO: 3.1.3

DESCRIPTION OF ISSUE: Add an exception to Condition C for LCO 3.0.4 since it should allow a MODE transfer.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 58

CHAPTER/LCO: 3.1.3

DESCRIPTION OF ISSUE: Revise LCO first sentence to read as in the NUREG.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 59

CHAPTER/LCO: 3.1.5

DESCRIPTION OF ISSUE: Revise LCO bases to add after first sentence "and prior to withdrawal of any control rod." This also impacts proposed WOG traveller.

DATE IDENTIFIED: 9/19/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 60

CHAPTER/LCO: 3.1.7

DESCRIPTION OF ISSUE: Verify accident analysis assumptions for maximum difference between most withdrawn and least withdrawn rod is 24 steps. This impacts Condition C.1.2 and SR 3.1.7.1. Revise bases accordingly. This affects change 15.iii.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: The LCO bases for LCO 3.1.4 state that a misalignment of 25 steps is assumed.

20-Dec-95

ITEM #: 61

CHAPTER/LCO: 3.1.7

DESCRIPTION OF ISSUE: Revise ACTIONS Note to be consistent with NUREG note since you will be in Condition A and E with more than 1 MRPI per group inoperable; therefore need separate Condition entry. This affects change 15.ix. Replace "rod position indicator" in Note with "MRPI".

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 62

CHAPTER/LCO: 3.2.3

DESCRIPTION OF ISSUE: Add a statement to the LCO Bases that "The COLR contains the figure showing the target band and AFD acceptable operation limits."

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Only added a statement for the target band since bases already discussed the COLR and AFD acceptable operation limits.

20-Dec-95

ITEM #: 63

CHAPTER/LCO: 3.2.3

DESCRIPTION OF ISSUE: Add a new SR to verify that AFD monitor alarm is operable every 12 hours. Add bases explanation concerning how current practice of using computer to generate a control band alarm satisfies this test.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Based on conversation with Control Room.

ITEM #: 64

CHAPTER/LCO: 3.1.1

DESCRIPTION OF ISSUE: Revise SR 3.1.1.1 to be on a frequency of every 24 hrs. instead of proposed 48 hours.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Current Ginna practice is 24 hours. Agreed to by D. Filion.

20-Dec-95

ITEM #: 65

CHAPTER/LCO: 3.2.3

DESCRIPTION OF ISSUE: Revise RA B.1 and its associated Completion Time to be consistent with NUREG, Rev. 1.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: If B.1 is not completed in time, LCO 3.0.3 becomes Applicable and shutdown initiated until the LCO is again met.

ITEM #: 66

CHAPTER/LCO: 3.2.3

DESCRIPTION OF ISSUE: Revise RA C.1 and its associated Completion Time to be consistent with NUREG, Rev. 1.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: If B.1 is not completed in time, LCO 3.0.3 becomes Applicable and shutdown initiated until the LCO is again met.

20-Dec-95

ITEM #: 67

CHAPTER/LCO: 3.2.4

DESCRIPTION OF ISSUE: Add a new SR to verify that the QPTR monitor alarm is operable every 12 hours. Add bases explanation concerning current practice.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: See comment #63

ITEM #: 68

CHAPTER/LCO: 3.2.4

DESCRIPTION OF ISSUE: Track new Traveller for LCO 3.2.4 and provide markups and justification for implementation.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME: TSTF-25

COMMENTS:

ITEM #: 69

CHAPTER/LCO: 3.2.4

DESCRIPTION OF ISSUE: Expand bases for RA C.1, C.2, and SR 3.2.4.2 to show that 24 hours is consistent with the 24 hours to do a flux map with an inoperable power range channel greater than or equal to 75 % RTP. Since power range channels feed the QPTR monitor, the Completion Times and Frequencies should all be consistent.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 70

CHAPTER/LCO: 3.1.6

DESCRIPTION OF ISSUE: Revise SR 3.1.6.1 to have the same Frequency as the NUREG. This impacts 3.1Q7.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by P. Bamford.

20-Dec-95

ITEM #: 71

CHAPTER/LCO: 3.5.1

DESCRIPTION OF ISSUE: Initiate a traveller to relocate accumulator boron concentrations to the COLR.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 2/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This traveller was rejected by the Reactor Systems branch. See comment #220.

ITEM #: 72

CHAPTER/LCO: 3.5.2

DESCRIPTION OF ISSUE: Provide additional justification why SR 3.5.2.8 should not be required.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/ 2/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This SR was added to the ITS such that no further justification is required.

ITEM #: 73

CHAPTER/LCO: 3.5.4

DESCRIPTION OF ISSUE: Determine how SFDP applies to an inoperable RWST.

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 10/23/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Tracked by #1.

ITEM #: 74

CHAPTER/LCO: 3.4.1

DESCRIPTION OF ISSUE: Add reference to performance of SR 3.4.4.1 in Section C change 32.iv.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary changes to Section D were made such that no further changes to Section C were required.

ITEM #: 75

CHAPTER/LCO: 3.4.1

DESCRIPTION OF ISSUE: Initiate traveller for changes to NUREG SR 3.4.1.4.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME: WOG-37

COMMENTS: Still undergoing industry review.

20-Dec-95

ITEM #: 76

CHAPTER/LCO: 3.4.2

DESCRIPTION OF ISSUE: Add a new SR to verify Tavg in each loop greater than or equal to 540F within 30 minutes prior to criticality.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Per M. Ruby and P. Bamford 30 minutes should be acceptable and P. Bamford.

ITEM #: 77

CHAPTER/LCO: 3.5.2

DESCRIPTION OF ISSUE: Revise insert 3.5.14 to only apply to valves 878B and 878D since 878A and 878C are never touched.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/ 2/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by G. Joss.

20-Dec-95

ITEM #: 78

CHAPTER/LCO: 3.4.4

DESCRIPTION OF ISSUE: Correct Attachment D Applicability to show "MODE 1, > 8.5% RTP."

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary corrections to Attachment B were made such that no further changes to Attachment D are required.

ITEM #: 79

CHAPTER/LCO: 3.4.5

DESCRIPTION OF ISSUE: Revise insert 3.4.11 to add "consistent with the safety analysis assumptions."

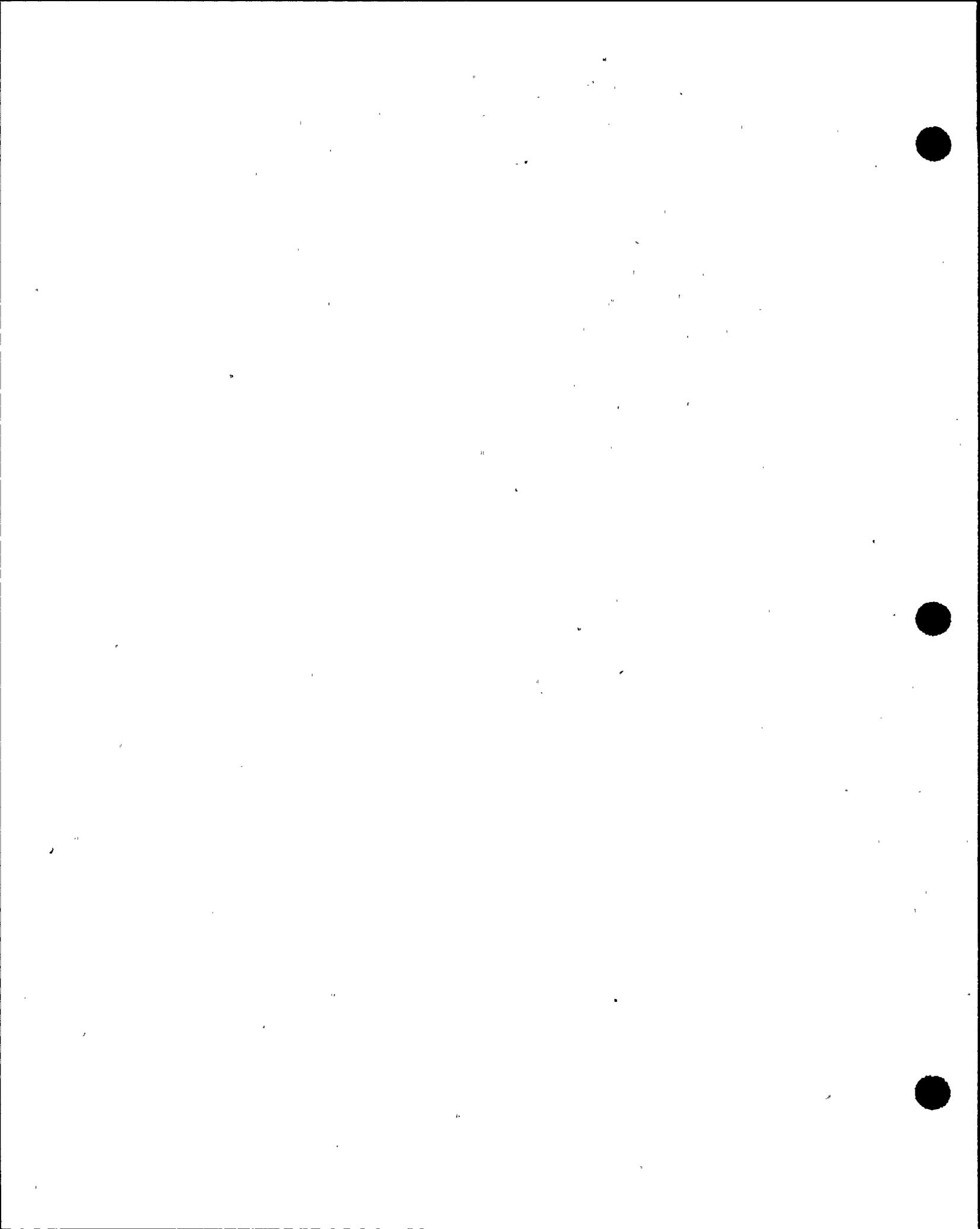
DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



ITEM #: 80

CHAPTER/LCO: 3.4.7

DESCRIPTION OF ISSUE: Change Condition B to read "Both RHR loops..." This change is also required for Condition B of LCO 3.4.8.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 81

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Revise Insert 3.4.47 to say "To maintain this pressure differential limit..."

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 82

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Add "ECCS" to the entire section when discussing "accumulators".

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 83

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Change "verify" to "ensure" in Required Action A.1 and B.1.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 84

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Reorder LCO conditions based on which option of LCO 3.4.12 is taken.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 85

CHAPTER/LCO: 3.4.14

DESCRIPTION OF ISSUE: Propose traveller for SR 3.4.14.1, Note 1 consistent with change 45.v.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #170.

20-Dec-95

ITEM #: 86

CHAPTER/LCO: 3.4.14

DESCRIPTION OF ISSUE: Provide a justification related to OMa-1988 on how a frequency of 24 months is acceptable for SR 3.4.14.1.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 87

CHAPTER/LCO: 3.4.15

DESCRIPTION OF ISSUE: Add discussion to Sectio C change 46.vi that you can only use the containment air cooler collection system if the radiation monitors are OPERABLE.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary changes were made to Section D such that no further changes to Section D are required.

20-Dec-95

ITEM #: 88

CHAPTER/LCO: 3.4.15

DESCRIPTION OF ISSUE: Add discussion to Section C change 46.iii to address the fact that with the required radiation monitor inoperable, we not have the option to perform an inventory balance every 24 hrs. in lieu of doing grab samples.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary changes were made to Section D such that no further changes to Section D are required.

ITEM #: 89

CHAPTER/LCO: 3.4.15

DESCRIPTION OF ISSUE: Delete SR 3.4.15.5 and add discussion to Insert 3.4.65 stating that "and a CHANNEL CALIBRATION of the monitor has been performed within the last 24 months."

DATE IDENTIFIED: 9/20/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 90

CHAPTER/LCO: 3.4.16

DESCRIPTION OF ISSUE: Revise Note to SR 3.4.16.3 to delete "only" from note. Must check all uses of "only" in the SRs. This closes comment 3.7Q3.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #170.

ITEM #: 91

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Expand change 25.vi (CTS) to discuss how LTOP requirements are in CTS 3.15 and 3.3 and how we have maintained the MODE of Applicability for CTS 3.3.

DATE IDENTIFIED: 9/22/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 92

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Correct typographical errors and provide minor clarifications to Attachment A of the submittal. This closes comments: 3.7Q2, 3.7Q38, 3.7Q66, 3.7Q114 and 3.7Q126.

DATE IDENTIFIED: 9/25/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 93

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Perform minor bases clarifications to address typographical errors, minor clarifications, and editorial preference. This closes comments: 3.7Q5, 3.7Q52, 3.7Q56, 3.7Q57, 3.7Q58, 3.7Q59, 3.7Q73, 3.7Q86, 3.7Q144, , 3.7Q164, 3.7Q170, and 3.7Q185.

DATE IDENTIFIED: 9/25/95

DATE CLOSED: 10/13/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 94

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Correct minor typographical errors within Attachment D of the submittal. This closes comments: 3.7Q26, 3.7Q32, 3.7Q147, 3.7Q162, 3.7Q175, and 3.7Q197.

DATE IDENTIFIED: 9/25/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary corrections to Attachment C were made such that no changes to Attachment D were required.

ITEM #: 95

CHAPTER/LCO: 3.7.3

DESCRIPTION OF ISSUE: Revise title to read "MFRV and Associated Bypass Valves and MFPDVs." This closes comment 3.7Q39 and 3.7Q45.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 96

CHAPTER/LCO: 3.7.4

DESCRIPTION OF ISSUE: Add NUREG SR 3.7.4.2 back into the ITS since the block valve is credited in the accident analysis.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 97

CHAPTER/LCO: 3.7.4

DESCRIPTION OF ISSUE: Revise Attachment A, Section C, change 79.iii to discuss revision of the Completion Time for ITS Required Action B.1. This closes comment 3.7Q61.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 10/29/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 98

CHAPTER/LCO: 3.7.5

DESCRIPTION OF ISSUE: Provide a justification in Attachment A, Section C, item 80 for the Frequency of AFW and SAFW pump testing. Also track WOG traveller on this issue. This closes comments 3.7Q67 and 3.7Q68.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME:

COMMENTS: Traveller approval being tracked by #99.

ITEM #: 99

CHAPTER/LCO: 3.7.5

DESCRIPTION OF ISSUE: Track Traveller with respect to elimination of MODE 4 requirements for AFW. This closes comments 3.7Q65, 3.7Q70, 3.7Q79, 3.7Q80, and 3.7Q87.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME: TSIF-29

COMMENTS: Submitted to NRC on 11/13/95.

20-Dec-95

ITEM #: 100

CHAPTER/LCO: 3.7.7

DESCRIPTION OF ISSUE: Add a note to Required Action C.1 similar to the note for LCO 3.7.5, Required Action D.1, which suspends implementation of LCO 3.0.3 and all MODE reductions. This closes comment 3.7Q95.

DATE IDENTIFIED: 9/26/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 101

CHAPTER/LCO: 3.7.7

DESCRIPTION OF ISSUE: Provide a drawing in the bases which shows the breakdown of the CCW pump trains and loop header. This closes comment 3.7Q99, 3.7Q102 and 3.7Q108.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 102

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Add a note to Required Action C.1 similar to the note for LCO 3.7.5, Required Action D.1, which suspends implementation of LCO 3.0.3 and all MODE reductions. This closes comments 3.7Q113, 3.7Q116, 3.7Q117, 3.7Q118, and 3.7Q124

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This change was rejected in meeting on 10/12/95 (#135). Therefore, change was closed.

ITEM #: 103

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Provide a drawing in the bases which shows the breakdown of the SW pump trains and loop header. This closes comment 3.7Q123.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 104

CHAPTER/LCO: 3.7.9

DESCRIPTION OF ISSUE: Provide a drawing to the bases which shows the breakdown of the CREATS filtration trains and dampers. This closes comment 3.7Q134.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 105

CHAPTER/LCO: 3.7.9

DESCRIPTION OF ISSUE: Change the logical connector between condition D.1 and D.2 to an OR and revise the bases accordingly. This closes comment 3.7Q138.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 106

CHAPTER/LCO: 3.7.10

DESCRIPTION OF ISSUE: Provide a drawing to the bases which shows what portion of the ABVs is addressed by this LCO. This closes comments 3.7Q155 and 3.7Q163.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 107

CHAPTER/LCO: 3.7.12

DESCRIPTION OF ISSUE: Request Traveller to relocate SFP boron concentration to the COLR. This closes comment 3.7Q182.

DATE IDENTIFIED: 9/28/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Reactor Systems Branch rejected change. Modified LCO to add concentration back in. See also #220 and #207.

20-Dec-95

ITEM #: 108

CHAPTER/LCO: 3.7.2

DESCRIPTION OF ISSUE: Track traveller for Completion Time for Required Action C.2. This closes comment 3.7Q18.

DATE IDENTIFIED: 9/28/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This comment closed by incorporation of #134. This traveller will not be implemented.

ITEM #: 109

CHAPTER/LCO: 3.9

DESCRIPTION OF ISSUE: Correct reference in Attachment B, CTS 3.5.5.1 to be "15.ix" instead of "15.viii" and provide new change justification 15.ix to Attachment A. This closes comment 3.9Q4.

DATE IDENTIFIED: 10/2/95

DATE CLOSED: 11/29/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This comment was superceded by #221.

ITEM #: 110

CHAPTER/LCO: 3.9

DESCRIPTION OF ISSUE: Provide minor clarifications to the LCO and basis. This closes comments 3.9Q19 and 3.9Q27.

DATE IDENTIFIED: 10/ 3/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 111

CHAPTER/LCO: 3.9.2

DESCRIPTION OF ISSUE: Revise Insert 3.9.7.b to be more specific with respect to CHANNEL CHECKS. Also, check to see if the revised Insert 3.9.7.b should be added to SR 3.3.1.1 and SR 3.4.15.1. This closes comment 3.9Q25.

DATE IDENTIFIED: 10/ 3/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This comment was revised per #195 to delete the associated text.

20-Dec-95

ITEM #: 112

CHAPTER/LCO: 3.9.2

DESCRIPTION OF ISSUE: Revise SR 3.9.3.1 and 3.9.4.1 to be consistent with WOG Traveller. Also, track status of this traveller. This closes comments 3.9Q32 and 3.9Q34.

DATE IDENTIFIED: 10/ 3/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 113

CHAPTER/LCO: 3.9.2

DESCRIPTION OF ISSUE: Correct typographical errors and other minor corrections to Attachment A. This closes comment 3.7Q32 and 3.9Q34.

DATE IDENTIFIED: 10/ 3/95

DATE CLOSED: 11/29/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary corrections to Attachment A related to CTS markup were made such that no changes to the ITS markup are required.

20-Dec-95

ITEM #: 114

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment B. This closes comment 3.7Q211.

DATE IDENTIFIED: 10/4/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 115

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Correct change category for 13.xix to "v.b" and provide no significant hazards evaluation. This closes comment 3.7Q220.

DATE IDENTIFIED: 9/27/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 116

CHAPTER/LCO: 3.1.3

DESCRIPTION OF ISSUE: Initiate a traveller to revise EOL MTC verification consistent with what Ginna submitted.

DATE IDENTIFIED: 9/21/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Withdrew since WOG program for a similar change is currently undergoing WOG review.

ITEM #: 117

CHAPTER/LCO: 3.4.11

DESCRIPTION OF ISSUE: Revise conditions for PORV block valves to allow 72 hours with 2 inoperable valves and 7 days for one valve. Cite difficulty in repairs, probability of tube rupture, etc. Reference justification in 9/15/92.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 118

CHAPTER/LCO: 3.6

DESCRIPTION OF ISSUE: Editorial comments in bases: pages B3.6-8, B3.6-7, B3.6-22, B3.6-39, B3.6-40, Insert 3.6.1.4, B3.6-44, Insert 3.6.6.7, Insert 3.6.7.3, B3.6-63, and B3.6-67.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 119

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Track Traveller associated with changes to SR 3.6.2.2.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME: TSTF-177

COMMENTS:



20-Dec-95

ITEM #: 120

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Develop white paper on crediting closed systems. Discuss: (1) PRA, (2) Outside valves not leak tested whereas closed systems have leakage verification, (3) Not subject to an active failure, (4) Propose reasonable time 7/14 day. Provide copies of examples. Revise LCO per 11/16/95 appeal meeting.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 121

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Develop white paper for LCO note excluding MSIVs/MSSVs/ARVs from LCO 3.6.3. Cite: (1) Hardship, (2) Inconsistency in NUREG. Revise LCO per 11/16/95 appeal meeting.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 122

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Replace use of "barrier" with "boundary" consistent with CTS language. Remarkup this LCO and bases using Rev. 1 of NUREG.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 123

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Clarify SR 3.6.3.1 and SR 3.6.3.2. Also, revise Frequencies to be 92 days for SR 3.6.3.1 and "Prior to entering Mode 4... 92 days" for SR 3.6.3.2.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 124

CHAPTER/LCO: 3.6

DESCRIPTION OF ISSUE: Make minor editorial changes to LCOs. This affects pages: 3.6-8, 3.6-9, 3.6-11, Insert 3.6.3.2. Insert 3.6.3.18, Insert 3.6.3.15, and 3.6-24.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 125

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Revise Actions Note 4 bases to clarify that the evaluation of LCO 3.6.1 leakage limits applies to any boundary that is declared inoperable (both one and two inoperable boundaries). Carl must address the timeline of this action.

DATE IDENTIFIED: 10/10/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #192.

20-Dec-95

ITEM #: 126

CHAPTER/LCO: 3.6.3

DESCRIPTION OF ISSUE: Add NUREG SR 3.6.3.2 back into TS with reference to "admin controls" versus specific reasons for opening. These specific reasons should be in the SR bases. Also, control board verification is acceptable for this SR.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 127

CHAPTER/LCO: 3.6.4

DESCRIPTION OF ISSUE: Revise completion time for A.1 to 8 hours. This closes comment 3.6Q61.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 128

CHAPTER/LCO: 3.6.5

DESCRIPTION OF ISSUE: Change Frequency for SR 3.6.5.1 to be 12 hours. This closes comment 3.6Q71. Also, revise response to 3.6Q70 to discuss total duration of LCO not being met is (12 + 12 hrs) vs. the NUREG (8 + 24 hrs) or only a 4 hr difference.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 129

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Prepare a white paper discussing how to handle single train systems and SFDP (RWST, CST, CREATS). Due by 11/1.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Cancelled per phone call with C. Grimes on 10/11/95. NRC stated to address generically. If resolution not reached prior to SER, clarify bases for LCO 3.0.6 to state that SFDP does not apply to RWST & CSR. NRC would not fight this interp.

20-Dec-95

ITEM #: 130

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Revise "spray additive tank" to be "NaOH System" throughout LCO and bases.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by Operations.

ITEM #: 131

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Add a new SR similar to SR 3.6.6A.3 to "verify service water flow through each CRFC unit" every 31 days.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 132

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Renumber SRs so that the VFTP requirements came right before those with 24 month surveillance test intervals.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 133

CHAPTER/LCO: 3.7.4

DESCRIPTION OF ISSUE: Add "line" back into the LCO for ARVs. This closes comment 3.6Q52.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 134

CHAPTER/LCO: 3.7.2

DESCRIPTION OF ISSUE: Revise 3.7.2 to be consistent with NUREG and decisions made during RG&E/NRC meeting.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 135

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Make minor editorial changes to LCOs. This affects pages: 3.7-7, 3.7-8, 3.7-20, Insert 3.7.8.1, 3.7-18, 3.7-24, 3.7-23, 3.7-31, 3.7-35, and 3.7-40.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 136

CHAPTER/LCO: 3.7.3

DESCRIPTION OF ISSUE: Add drawing of MFW isolation valves to bases.

DATE IDENTIFIED: 10/11/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 137

CHAPTER/LCO: 3.7

DESCRIPTION OF ISSUE: Make minor editorial changes to bases. This affects pages: Insert 3.7.5.10, B3.7-26, B3.7-28, Insert 3.7.8.9, and 3.7-52

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 138

CHAPTER/LCO: 3.7.5

DESCRIPTION OF ISSUE: Provide additional justification to response to 3.7Q261 using: (1) SI/RHR/CS test frequencies, (2) pumps used more often during low power, (3) 5 pumps total, (4) reliability data (?). Also generate a traveller.

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Addressed by #98.

ITEM #: 139

CHAPTER/LCO: 3.7.5

DESCRIPTION OF ISSUE: Revise LCO based on issues discussed during RG&E/NRC meeting.

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 140

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Do a search for all plural words in the conditions and how it matches to Required Actions. Use Writer's Guide example to standardize.

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 141

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Add a new SR for "verify all SW loop header cross-tie valves are locked and in the correct position" every 31 days. Must verify 4665, 4760, 4669, 4668B, 4623, 4756, 4639, and 4640 to be locked open. Also verify 4612, 4611, 4610, and 4779 to be locked closed.

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 142

CHAPTER/LCO: 3.7.8

DESCRIPTION OF ISSUE: Add 1 new SR as follows: "Verify screenhouse bay water level and temperature are within limits "every 24 hours.

DATE IDENTIFIED: 10/12/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 143

CHAPTER/LCO: 3.7.10

DESCRIPTION OF ISSUE: Expand ASA bases to discuss why the remainder of the ABVS is not required for this LCO.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 144

CHAPTER/LCO: 3.7.10

DESCRIPTION OF ISSUE: Add a new SR to "verify the ABVS maintains a negative pressure with respect to atmosphere for the Auxiliary Building operating floor level" every 24 hours.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 10/27/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 145

CHAPTER/LCO: 3.7.12

DESCRIPTION OF ISSUE: Generate traveller to relocate SFP boron concentrations to COLR.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 10/13/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Addressed by #107.

20-Dec-95

ITEM #: 146

CHAPTER/LCO: 3.7.12

DESCRIPTION OF ISSUE: Expand response to 3.7Q191 to discuss: (1) need to drop 2000 - 300 ppm (time), (2) boraflex issues, (3) level being verified every 7 days.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 10/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 147

CHAPTER/LCO: 3.7.13

DESCRIPTION OF ISSUE: Generate traveller to relocate specification 4.3.1.1 operational details (e and f) to LCO 3.7.17. Also, revise LCO, item a, to delete all wording after "1.458" as this is bases information.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Traveller due from WOG in Dec.



20-Dec-95

ITEM #: 148

CHAPTER/LCO: 3.7.11

DESCRIPTION OF ISSUE: Propose a new traveller to add a LCO 3.0.4 exclusion to LCO 3.7.11, 3.7.12, and 3.7.13 similar to the LCO 3.0.3 exclusion. Add this to the ITS.

DATE IDENTIFIED: 10/13/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Rejected by WOG on 11/8/95 since actions are "immediate". Therefore, this was not added.

ITEM #: 149

CHAPTER/LCO: .5.5.10

DESCRIPTION OF ISSUE: Add the following to the last sentence of the bases for SR 3.6.6.5, SR 3.6.6.6, SR 3.7.9.2, and SR 3.7.10.2, "However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles defined by Regulatory Guide 1.52." This closes commitment in 10/18/95 letter to NRC.

DATE IDENTIFIED: 10/18/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 150

CHAPTER/LCO: 5.5.10

DESCRIPTION OF ISSUE: Add discussion on minimum required flowrates to the bases for SR 3.6.6.5, SR 3.6.6.6, SR 3.7.9.2, and SR 3.7.10.2. This closes commitment in 10/18/95 letter to NRC.

DATE IDENTIFIED: 10/18/95

DATE CLOSED: 10/31/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 151

CHAPTER/LCO: 5.5.8

DESCRIPTION OF ISSUE: Remove change 120.xvii and the addition of high energy piping and the steam generator tubes from the ITS Program. This closes comment 6.0Q6.

DATE IDENTIFIED: 10/18/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 152

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Replace all uses of "absorber" with "adsorber." This closes comment 6.0Q12.

DATE IDENTIFIED: 10/18/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 153

CHAPTER/LCO: 5.2.1

DESCRIPTION OF ISSUE: Add NUREG Section 5.2.1.d back into the submittal and revise change 50.iii. This closes comment 6.0Q27.

DATE IDENTIFIED: 10/19/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 154

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Correct typographical errors within Attachment D of the submittal. This closes comment 6.0Q33.

DATE IDENTIFIED: 10/19/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary corrections to Attachment C were made such that no changes to Attachment D were required.

ITEM #: 155

CHAPTER/LCO: 5.5.12

DESCRIPTION OF ISSUE: Revise diesel fuel oil testing proram to only require testing of viscosity, water, and sediment. This closes comment 6.0Q36.

DATE IDENTIFIED: 10/19/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Comment was replaced by #196.

20-Dec-95

ITEM #: 156

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Perform minor editorial changes to ITS 5.5.5 and 5.5.10. This closes comment 6.0Q37.

DATE IDENTIFIED: 10/19/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 157

CHAPTER/LCO: 1.0

DESCRIPTION OF ISSUE: Correct typographical errors and editorial changes to Attachment A of the submittal. This closes comments 1.0Q2 and 1.0Q13.

DATE IDENTIFIED: 10/20/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 158

CHAPTER/LCO: 1.1

DESCRIPTION OF ISSUE: Initiate and track a traveller to add "time constant" to definition of CHANNEL CALIBRATION. This closes comment 1.0Q8.

DATE IDENTIFIED: 10/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: RG&E withdrew use of time constants in definition.

ITEM #: 159

CHAPTER/LCO: 1.3

DESCRIPTION OF ISSUE: Revise change 3.vii to remove use of "either" and replace with NUREG use of "one" and to delete added text to page 1.3-11. This closes comment 1.0Q27.

DATE IDENTIFIED: 10/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 160

CHAPTER/LCO: 1.3

DESCRIPTION OF ISSUE: Revise ITS description to delete change 3.ix. This closes comment 1.0Q25.

DATE IDENTIFIED: 10/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 161

CHAPTER/LCO: 1.3

DESCRIPTION OF ISSUE: Revise location of Insert 1.3.1 to be right after first sentence at top of page 1.3-2. This closes comment 1.0Q26.

DATE IDENTIFIED: 10/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



ITEM #: 162

CHAPTER/LCO: 3.0

DESCRIPTION OF ISSUE: Remove text proposed at the end of the bases for SR 3.0.1 as change 7.xiii. This closes comment 3.0Q2.

DATE IDENTIFIED: 10/23/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 163

CHAPTER/LCO: 3.0

DESCRIPTION OF ISSUE: Generate and track a traveller to delete text clarifying "exceptions to the specification are provided in the individual specifications" for LCO 3.0.4. This change provides consistency with LCO 3.0.3 and closes comment 3.0Q7.

DATE IDENTIFIED: 10/23/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: Yes

TRAVELLER NAME:

COMMENTS: Traveller accepted by WOG on 11/8/95. Bases text revised consistent with proposed traveller.

20-Dec-95

ITEM #: 164

CHAPTER/LCO: 3.0

DESCRIPTION OF ISSUE: Track final WOG disposition of Traveller BWR-26, C.1. This closes comment 3.0Q7.

DATE IDENTIFIED: 10/23/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: NRC accepted proposed wording week of 11/20/95.

ITEM #: 165

CHAPTER/LCO: 3.0

DESCRIPTION OF ISSUE: Revise ITS and NUREG markup to retain deleted text in the second paragraph on page B 3.0-6. This closes comment 3.0Q7.

DATE IDENTIFIED: 10/23/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 166

CHAPTER/LCO: 4.0

DESCRIPTION OF ISSUE: Revise A-25.6 to include discussion of CTS 5.3.1.a reporting requirements for reconstituted fuel. This closes comment 4.0Q5.

DATE IDENTIFIED: 10/27/95

DATE CLOSED:

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Implementation issue only.

ITEM #: 167

CHAPTER/LCO: 4.0

DESCRIPTION OF ISSUE: Add CTS 5.4.4.5 into ITS 4.3.1.1 and correct reference in ITS 4.3.1.1.c to be Specification 3.7.13, not 3.7.17. This closes comment 4.0Q11.

DATE IDENTIFIED: 10/27/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 168

CHAPTER/LCO: 4.0

DESCRIPTION OF ISSUE: Generate and track a traveller to relocated NUREG 1431 4.3.1.1.e and 4.3.1.1.f to LCO 3.7.17. This closes comment 4.0Q16.

DATE IDENTIFIED: 10/27/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Traveller due in December.

ITEM #: 169

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Editorial comments within re-typed version (proof reading issues).

DATE IDENTIFIED: 10/28/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 170

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Update use of "only" within SR Notes to be consistent within NUREG.

DATE IDENTIFIED: 10/30/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 171

CHAPTER/LCO: 3.2.1

DESCRIPTION OF ISSUE: Split SR 3.2.1.1 (and SR 3.2.2.1) into 2 separate SRs consistent with rest of NUREG.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/ 6/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This was not incorporated since LCO 3.2.4 and SR 3.2.4.2 and SR 3.2.4.3 directly reference these SRs.

ITEM #: 172

CHAPTER/LCO: 3.4.12

DESCRIPTION OF ISSUE: Revise Frequency of SR 3.4.12.3 to be "once within 12 hours and every 12 hours thereafter." Also revise Frequency to be "once within 12 hours and every 31 days thereafter."

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 173

CHAPTER/LCO: 3.4.13

DESCRIPTION OF ISSUE: Revise SR 3.4.13.1 to be consistent with TSTF traveller

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Traveller was never submitted by the WOG such that change was left as is.

ITEM #: 174

CHAPTER/LCO: 3.4.16

DESCRIPTION OF ISSUE: Revise SR 3.4.16.3 to relocate all text after Note text "only required to be performed in MODE 1" to the Frequency column.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 175

CHAPTER/LCO: 3.7.1

DESCRIPTION OF ISSUE: Revise SR 3.7.1.1 Note to read "only required to be performed in MODES 1 and 2."

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 176

CHAPTER/LCO: 3.7.5

DESCRIPTION OF ISSUE: Revise notes for SR 3.7.5.2 and SR 3.7.5.6 to read "Required to be met prior to..."

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 177

CHAPTER/LCO: 3.5.3

DESCRIPTION OF ISSUE: Revise bases to state: 1. There are no ASA for this LCO, 2. The SI system must be isolated by LTOP by 2 independent means, 3. LCO being provided for "good operational practice.", 4. 10 minute actuation provided for LTOP protection and consistent with licensed bases for acutation., 5. LCO meets criterion 4, delete reference to WCAP.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/ 2/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: See B3.3-36 of MWR/4

ITEM #: 178

CHAPTER/LCO: 3.6.6

DESCRIPTION OF ISSUE: Add a new SR to 3.6.6 which states "Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B" in accordance with applicable SRs.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 179

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Add discussion of 40CFR141 and 40CFR190 from CTS 3.9.1.2.b (iii) to change justification 19.ii.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 180

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Revise 5.5.10 to: (1) Delete reference to ESF; (2) revise title of 5.5.10a to add "containment"; (3) add "at a design" prior to flow rate under 5.5.10.6.3, and (4) discuss 24 month frequency versus 18 month test frequency.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 181

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Add a CTS change justification to show where CTS 6.8.1.c went to.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/10/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 182

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Change the following CTS justifications to (iii): 56.ii, 57.vii, 64.i. Change the following CTS justifications to (ii): 57.vii. Change the following CTS justification to (v.a): 63.i. This closes meeting comments from week of 11/1/95.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/10/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 183

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Generate a matrix which shows: (1) all change category (ii) and what regulation duplicates it. (2) all change category (iii) and what regulation covers changes to it.

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 12/20/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 184

CHAPTER/LCO: 5.2.1

DESCRIPTION OF ISSUE: Revise 5.2.1.a to address use of lowercase letter titles consistent with 11/10/94 NRC letter. Change 5.2.1.b to read "The plant manager shall report to the corporate vice president specified in 5.2.1.c, and ..". Also, revise 5.2.1.c to read "A corporate vice president..."

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 185

CHAPTER/LCO: 5.7.1

DESCRIPTION OF ISSUE: Place "Radiation Protection Technician" in lower case letters. Also, add to 5.7.1.c as follows: "...at a frequency specified by the radiation protection technician in the RWP."

DATE IDENTIFIED: 11/ 1/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 186

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Renumber Secondary Chemistry Program (ITS 5.5.15) to come right after SG Tube Surveillance Program per NUREG.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 187

CHAPTER/LCO: 5.5.14

DESCRIPTION OF ISSUE: Remove changes justified by 121.iii from Attachment C of the submittal.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: NRC agrees with interpretation provided by the change but due to its generic implications, agreed to withdraw it.

ITEM #: 188

CHAPTER/LCO: 3.3.3

DESCRIPTION OF ISSUE: Replace use of "Special Report" with "report" in the ACTIONS and bases.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Used lower case "special report" per agreements during meetins week of 11/13/95.

ITEM #: 189

CHAPTER/LCO: 3.7.7

DESCRIPTION OF ISSUE: Revise LCO to require both CCW HXs to be OPERABLE and remove from loop definition. Allow one Hx to be inoperable for 31 days. Also require one Hx to be in operation and both OPERABLE. A CCW Hx which is isolated, but capable of being opened within accident analysis time limits is OPERABLE.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by J. Cook (RG&E) and Bill LeFave (NRC). Carl must review and approve also.

ITEM #: 190

CHAPTER/LCO: 5.2.2

DESCRIPTION OF ISSUE: Add a new requirement that the "Operations Manager or at least one operations middle manager shall hold an SRO" similar to NUREG and 11/10/94 NRC letter.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by Widay and Marchionda

20-Dec-95

ITEM #: 191

CHAPTER/LCO: 3.9.4

DESCRIPTION OF ISSUE: Add a CTS discussion to Attachment A related to the addition of new SR 3.9.4.2.

DATE IDENTIFIED: 11/2/95

DATE CLOSED: 11/29/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 192

CHAPTER/LCO: 3.6.2

DESCRIPTION OF ISSUE: Revise bases for Note 3 (and Note 4 for LCO 3.6.3) to state that action should be initiated immediately to verify CNMT is still OPERABLE upon declaring a boundary or airlock inoperable.

DATE IDENTIFIED: 11/2/95

DATE CLOSED: 11/30/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 193

CHAPTER/LCO: 5.5.10

DESCRIPTION OF ISSUE: Delete "where practical" in last sentence of specification.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by Joss.

ITEM #: 194

CHAPTER/LCO: 3.9

DESCRIPTION OF ISSUE: Make bases changes. This affects pages: B3.9-1, Insert 3.9.2, Insert 3.9.3, B3.9-3, B3.9-2, B3.9-4, B3.9-17, Insert 3.9.7b, Insert 3.9.6a, B3.9-9, Insert 3.9.7a, Insert 3.9.8, B3.9-18, B3.9-20, and B3.9-22. This closes comments generated during meetings week of 11/20/95.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Insert 3.9.7b and 3.9.6.a are correct but NRC prefers be added after their SER via 50.59.

20-Dec-95

ITEM #: 195

CHAPTER/LCO: 3.9

DESCRIPTION OF ISSUE: Make LCO changes. This affects pages: 3.9.1, 3.9.4, Insert 3.9.1.a, 3.9-5, 3.9-9, 3.9-10, 3.9-8, and 3.8-11. This closes comments generated during meetings week of 11/20/95.

DATE IDENTIFIED: 11/ 2/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 196

CHAPTER/LCO: 5.5.12

DESCRIPTION OF ISSUE: Revise 5.5.12 to include the NUREG items a (1-3) and the revised item b only. This effectively deletes the CTS fuel oil testing requirements for testing every 92 days. Initiate traveller for new b.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by Joss and Ed Tomlinson.

ITEM #: 197

CHAPTER/LCO: 2.0

DESCRIPTION OF ISSUE: Revise CTS justification 2.i to remove discussion of "subcritical in MODE 2." This closes comment 2.0Q1.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 198

CHAPTER/LCO: 2.0

DESCRIPTION OF ISSUE: Revise the following bases pages: B2.0-9 and B2.0-10. This closes meeting comments from week of 11/1/95.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 199

CHAPTER/LCO: 3.0

DESCRIPTION OF ISSUE: Revise the following bases page: B3.0-6. This closes meeting comments from week of 11/1/95.

DATE IDENTIFIED: 11/ 3/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 200

CHAPTER/LCO: 5.5.5

DESCRIPTION OF ISSUE: Revise text to reference specifically UFSAR Table 5.1-4.

DATE IDENTIFIED: 11/10/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: NRC agreed that even though this table is specifically referenced in the Admin Control TS, the table can still be revised under 50.59.



ITEM #: 201

CHAPTER/LCO: 5.5.7

DESCRIPTION OF ISSUE: Relocate Specification 5.5.7, RCP Flywheel Inspection to ISI Program since this program is not in the CTS and is expected to be relocated/deleted in early 1996 via a recently submitted WCAP.

DATE IDENTIFIED: 11/10/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 202

CHAPTER/LCO: 3.0.4

DESCRIPTION OF ISSUE: Revise bases for LCO 3.0.4 to add in discussion concerning why 3.0.4 applies in MODES 5 and 6.

DATE IDENTIFIED: 11/10/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: This is being tracked by #164.

20-Dec-95

ITEM #: 203

CHAPTER/LCO: 5.7.1

DESCRIPTION OF ISSUE: Revise 5.7.1.c to specifically state that the radiator protection technician signs the RWP similiar to NUREG-1431.

DATE IDENTIFIED: 11/10/95

DATE CLOSED: 11/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed by #185.

ITEM #: 204

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Add "For Illustation Only" to all drawings being added to the bases.

DATE IDENTIFIED: 11/10/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 205

CHAPTER/LCO: 5.6

DESCRIPTION OF ISSUE: Revise COLR and PTLR references to be consistent with final approved sources.

DATE IDENTIFIED: 11/11/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 206

CHAPTER/LCO: all

DESCRIPTION OF ISSUE: Provide additional justification for 62.viii and 80.ix which relates that 96% of the time these systems were fully in service and that 4% of the time, one train of a 4 train system was OOS. Provide a separate letter committing to implement Maint. Rule tracking on a LCO basis by June 1, 1996 implementation date.

DATE IDENTIFIED: 11/13/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: All necessary changes to Attachment B were made such that no further changes to Section C of Attachment A are required.

ITEM #: 207

CHAPTER/LCO: 3.7.12

DESCRIPTION OF ISSUE: Relocate SFP boron concentration back into LCO (vs. COLR).

DATE IDENTIFIED: 11/13/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 208

CHAPTER/LCO: 3.3.1

DESCRIPTION OF ISSUE: Relocate RTS OPDT and OTDT parameters back into LCO Table (vs. COLR)

DATE IDENTIFIED: 11/13/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Closed per #220.

ITEM #: 209

CHAPTER/LCO: 3.4.11

DESCRIPTION OF ISSUE: Withdraw RG&E resubmittal of PORV TS amendment.

DATE IDENTIFIED: 11/13/95

DATE CLOSED: 12/20/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Address in final submittal cover letter.



20-Dec-95

ITEM #: 210

CHAPTER/LCO: 3.4.11

DESCRIPTION OF ISSUE: Revise LCO per meeting discussions week of 11/1/95.

DATE IDENTIFIED: 11/13/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: NRC accepted LCo as is with only changes closed by #209 & 237.

ITEM #: 211

CHAPTER/LCO: 3.8.1

DESCRIPTION OF ISSUE: Revise change 17.iii to relocate second offsite power line requirements to TRM.

DATE IDENTIFIED: 11/14/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 212

CHAPTER/LCO: 3.8

DESCRIPTION OF ISSUE: Revise LCO pages: 3.8-2, 3.8-7, 3.8-8, 3.8-15, 3.8-16, 3.8-1, 3.8-18, 3.8-19, 3.8-4, 3.8-21, 3.8-22, 3.8-23, 3.8-28, 3.8-36, 3.8-38, 3.8-40, 3.8-41, 3.8-30, 3.8-31, 3.8-32, and 3.8-33.

DATE IDENTIFIED: 11/14/95

DATE CLOSED: 12/ 4/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 213

CHAPTER/LCO: 3.8

DESCRIPTION OF ISSUE: Revise bases pages: B3.8-3, B3.8-7, Insert 3.8.9.4

DATE IDENTIFIED: 11/14/95

DATE CLOSED: 12/ 4/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 214

CHAPTER/LCO: 3.8

DESCRIPTION OF ISSUE: Revise change 17.vi to be a "v.b" change.

DATE IDENTIFIED: 11/14/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 215

CHAPTER/LCO: 3.8.7

DESCRIPTION OF ISSUE: Ensure loss of Instrument Bus D is addressed in SFDP.

DATE IDENTIFIED: 11/14/95

DATE CLOSED:

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Implementation issue only.

ITEM #: 216

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Revise LCO pages: 3.3-44, 3.3-45, 3.3-46, 3.3-52, 3.3-54 per meeting agreements week of 11/13/95.

DATE IDENTIFIED: 11/15/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 217

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Revise Bases pages: B3.3-64

DATE IDENTIFIED: 11/15/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 218

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Revise LCO 3.3.2 per meeting agreements week of 11/13/95.

DATE IDENTIFIED: 11/16/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 219

CHAPTER/LCO: 2.1.1

DESCRIPTION OF ISSUE: Revise bases for ASA to retain high pressurizer pressure trip and low pressurizer pressure trip.

DATE IDENTIFIED: 11/18/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



20-Dec-95

ITEM #: 220

CHAPTER/LCO: 5.0

DESCRIPTION OF ISSUE: Revise COLR Admin Controls and all affected LCOs to place back within TS the following parameters: (1) RTS instrumentation, (2) accumulator boron concentration, (3) RWST boron concentration, and (4) SFP boron concentration.

DATE IDENTIFIED: 11/18/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 221

CHAPTER/LCO: 3.9

DESCRIPTION OF ISSUE: Add back into TS the LCOs for containment isolation and associated instrument per results of 11/16/95 appeal.

DATE IDENTIFIED: 11/18/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 222

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Revise LCO 3.3.1 per meeting agreements week of 11/13/95.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 223

CHAPTER/LCO: 3.3.2

DESCRIPTION OF ISSUE: Add AFW manual initiation back into TS per meeting agreements week of 11/13/95.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 224

CHAPTER/LCO: 3.9.2

DESCRIPTION OF ISSUE: Remove change 106.iii to Condition B.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 225

CHAPTER/LCO: 4.1

DESCRIPTION OF ISSUE: Revise specification to include UFSAR Table 2.3-26 for description of EAB..

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 226

CHAPTER/LCO: 4.0

DESCRIPTION OF ISSUE: Make changes to specifications 4.2.1, 4.3.1.1.b, 4.3.1.2.b, Insert 4.3.1.c, 4.3.1.2.c, 4.3.1.1.c, and 4.3.3, 4.2.2, and 4.3.2. This closes comments from meetings week of 11/20/95.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 11/28/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 227

CHAPTER/LCO: 1.1

DESCRIPTION OF ISSUE: Revise definitions for : CHANNEL CAL, COT, and TADOT to add "display" back in.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 228

CHAPTER/LCO: 1.0

DESCRIPTION OF ISSUE: Make various changes to specifications pages 1.2-1, 1.4-2

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 229

CHAPTER/LCO: 1.0

DESCRIPTION OF ISSUE: Revise markup for 1.8 to show that it is going to bases for LCO 3.6.1 and 3.6.2.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 230

CHAPTER/LCO: All

DESCRIPTION OF ISSUE: Review all instrumentation logic testing requirements to add ACTUATION LOGIC TESTS as needed.

DATE IDENTIFIED: 11/20/95

DATE CLOSED: 12/11/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 231

CHAPTER/LCO: 3.0.3

DESCRIPTION OF ISSUE: Revise LCO to remove 1 hour preparation time and remove all restrictions on when shutdown must begin. Complete - 12/1/95.

DATE IDENTIFIED: 12/ 1/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by White and Marchionda.

ITEM #: 232

CHAPTER/LCO: 3.0.1

DESCRIPTION OF ISSUE: Remove TSTF-08 change with respect to SR 3.0.1. This also impacts Chapter 3.8. Complete 12/8/95.

DATE IDENTIFIED: 12/ 1/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 233

CHAPTER/LCO: 1.1

DESCRIPTION OF ISSUE: Remove TSTF-03 change with respect to DOSE EQUIVALENT I-131 dose conversion factors. Add reference to RG 1.109. Complete 12/1/95

DATE IDENTIFIED: 12/ 1/95

DATE CLOSED: 12/ 1/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 234

CHAPTER/LCO: 3.8.4

DESCRIPTION OF ISSUE: Revise SR 3.8.4-1 to be consistent with NUREG-1431. This is necessary since recent mods have removed large loads from the batteries such that the proposed test would be extremely difficult. Complete 12/8/95.

DATE IDENTIFIED: 12/ 8/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:



ITEM #: 235

CHAPTER/LCO: 3.3.3

DESCRIPTION OF ISSUE: Revise LCO per meetings agreements week of 11/13/95. Complete 12/8/95

DATE IDENTIFIED: 12/ 8/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 236

CHAPTER/LCO: 3.3

DESCRIPTION OF ISSUE: Revise LCO 3.3.2, 3.3.4, and 3.3.5 to include all necessary ACTUATION LOGIC TESTS.

DATE IDENTIFIED: 12/ 8/95

DATE CLOSED: 12/ 8/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS: Agreed to by G. Joss.

20-Dec-95

ITEM #: 237

CHAPTER/LCO: 3.4.11

DESCRIPTION OF ISSUE: Revised note for 3.4.11.1 consistent with discussions with Plant System's Branch.

DATE IDENTIFIED: 12/12/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 238

CHAPTER/LCO: 3.4.16

DESCRIPTION OF ISSUE: Replace Figure 3.4.16-1 with the new Figure based on NRC conference call of 11/27/95.

DATE IDENTIFIED: 12/12/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 239

CHAPTER/LCO: 3.4.13

DESCRIPTION OF ISSUE: Remove proposed Condition B since the WOG has decided not to pursue this issue. Revert back to the NUREG, Rev. 1.

DATE IDENTIFIED: 12/12/95

DATE CLOSED: 12/12/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 240

CHAPTER/LCO: 5.2.2

DESCRIPTION OF ISSUE: Revise to read "contained in the STA training program specified in UFSAR Section 13.2."

DATE IDENTIFIED: 12/14/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

20-Dec-95

ITEM #: 241

CHAPTER/LCO: 5.2.2

DESCRIPTION OF ISSUE: Revise 5.2.2.d to read "in accordance with a NRC approved program specified in plant procedures. Changes to the guidelines in these procedures shall be submitted to the NRC for review."

DATE IDENTIFIED: 12/14/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 242

CHAPTER/LCO: 5.5.4

DESCRIPTION OF ISSUE: Revise 5.5.4.d to add "and 40 CFR 141 Safe Drinking Water Act" to end of sentence.

DATE IDENTIFIED: 12/14/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 243

CHAPTER/LCO: 5.5.6

DESCRIPTION OF ISSUE: Revise the Tendon Surveillance Program to state "in accordance with Regulatory Guide 1.35, Revision 2."

DATE IDENTIFIED: 12/14/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

ITEM #: 244

CHAPTER/LCO: 5.5.7

DESCRIPTION OF ISSUE: Revise the first paragraph to read "in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g)."

DATE IDENTIFIED: 12/14/95

DATE CLOSED: 12/19/95

TRAVELLER EXIST?: No

TRAVELLER NAME:

COMMENTS:

Attachment K

Response to NRC Questions Contained in Letter Dated December 1, 1995

December 1995

1. Technical Specification 1.0

- i. TS 1.2 - The definitions of operating MODES were revised as follows (these are Ginna TS Category (v.a) changes):
- a. Refueling - see Note 1.ii below.
 - b. Cold Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits.

1.0Q1 Document the conclusion that the ITS reactivity limit is equivalent to the CTS requirements. Note that administrative changes are described as follows: Non-technical, administrative changes are intended to incorporate human-factors principles into the form and structure of the improved plant TS so that they would be easier to use for plant operations personnel. These changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content or operational requirements.

Response: Defining reactivity limits in form of k_{eff} in the ITS is actually more conservative than $\Delta k/k\%$ as is done in the CTS. Reactivity is defined as $(k_{eff} - 1)/k_{eff}$ (i.e., $\Delta k/k$). Placing a k_{eff} value of 0.99 in this equation yields a value of $-0.010101 \Delta k/k$ or $-1.0101 \Delta k/k\%$ which is more conservative than the CTS value of $-1 \Delta k/k\%$. For refueling, a k_{eff} value of 0.95 yields a reactivity of $-0.05263 \Delta k/k$ or $-5.26 \Delta k/k\%$. This is again more conservative. Therefore, the larger the shutdown margin required in ITS, the more conservative the use of k_{eff} in defining reactivity becomes. However, this is still considered an administrative change since the ITS specifies the input into the reactivity equation while CTS specifies the output.

- c. Hot Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits. The average reactor coolant temperature was also revised from $\geq 540^\circ\text{F}$ to $\geq 350^\circ\text{F}$. This change eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2.

1.0Q2 Does the statement, "This change eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2." mean that the 350 degree number was used to establish a mode in the ITS that is equivalent to the CTS mode limits? Explain. The expansion of this temperature range is conservative since the current TS only use the Hot Shutdown MODE in two aspects. The first method is requiring a shutdown to this mode due to plant conditions. Since the upper temperature range for Hot Shutdown remains the same (i.e., the Operating MODE temperature), there is no impact.

Response: The CTS use a temperature of 350°F to establish a specified condition or mode in multiple specifications. However, this

temperature is not specifically denoted in the CTS Mode definitions (i.e., this temperature limit was "backfitted" into the CTS following TMI without revising CTS 1.2). This specified condition is equivalent to the ITS operational MODE separation between Hot Shutdown and Hot Standby in Table 1.1-1. Any differences in the utilization of the defined mode in the ITS from any CTS requirement is annotated as a technical change and discussed and justified in their applicable sections. Therefore, this change to CTS 1.2 is merely an administrative presentation preference.

For the CTS which do not include a specified condition of 350°F (e.g., require a system to be OPERABLE when RCS > 200°F), the CTS have been revised to clarify that a MODE change occurs at 350°F. For these conditions, the ITS requirement would be more restrictive than the corresponding CTS requirement due to LCO 3.0.4 issues.

The intent of the above changes is to provide clarity and completeness in avoiding any misinterpretation, and could be considered administrative. However, since this change also eliminates the potential to interpret certain plant conditions such that no mode, or less restrictive mode would exist, this change should be modified to reflect that it is actually a more restrictive change. Comment #157 has been opened to add a more restrictive notation to this item.

1.0Q3

Clarify the statement " Since the upper temperature range for Hot Shutdown remains the same (i.e., the Operating MODE temperature), there is no impact." Is there zero impact or is the impact inconsequential? Explain. The second method is to require certain equipment to be OPERABLE in this mode. However, lowering the temperature limit to 350°F requires that the equipment would be OPERABLE for a larger temperature range.

Response: *There is no impact of maintaining the upper temperature range the same in ITS and CTS even though the upper range is only specified in ITS LCO 3.4.1 and not in CTS. See also response to 1.0Q2.*

d. Operating - The reactivity limit was revised from $> -1 \Delta k/k\%$ to $\geq 0.99 k_{\text{eff}}$ which are equivalent limits. The average reactor coolant temperature of $\sim 580^\circ\text{F}$ was not added since this parameter is specified in new LCO 3.4.1.

1.0Q4

LCO 3.4.1 establishes DNBR limits. Clarify how the temperature limit for existing TS reactivity limits are provided for in the ITS. In addition, the Operating MODE was separated into two modes: Operating and Startup. The only difference between these two modes is that Startup is defined when the reactor is $\leq 5\%$ Rated Thermal Power (RTP) while the Operating MODE is when the reactor is $> 5\%$ RTP.

Response: *The CTS Mode definition "Operating" specifies an average reactor coolant temperature of "approximately 580°F." The ITS Mode definition Table 1.1-1 does not specify a temperature. The ITS for*

Modes 1 and 2 is designated as "NA" on the basis that temperature for these modes is dictated by LCO 3.4.1 and LCO 3.4.2 (minimum temperature for criticality). The significance of limiting the RCS temperature is to ensure that the DNB design criterion is met for each of the transients analyzed. The average reactor coolant temperature as specified in LCO 3.4.1 is consistent with operation within the nominal operation envelope as assumed in the accident analyses. Since the average reactor coolant temperature is limited at the upper end by LCO 3.4.1 and the lower temperature for these modes are dictated by LCO 3.4.2, this change is merely an administrative presentation preference.

1.0Q5 The statement about the difference between the operating and startup modes is factual. Explain why this is or is not a change to the existing operational requirements for Ginna? Provide supporting justification for your response.

Response: This change is merely an administrative presentation preference similar to changes discussed in the response to 1.0Q2. However, it could also be considered a more restrictive change to the CTS since a MODE change now occurs at 5% RTP where none previously occurred (i.e., LCO 3.0.4 issues).

e. A new operating mode (Hot Standby) was provided between Hot Shutdown and Cold Shutdown. This mode is defined as when the reactivity condition is $< 0.99 k_{eff}$ and the average reactor coolant temperature is $< 350^{\circ}\text{F}$ and $> 200^{\circ}\text{F}$ when the reactor vessel head closure bolts are fully tensioned. The definition of this new mode eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2.

1.0Q6 Explain why this is or is not a change to the existing operational requirements for Ginna? Provide supporting justification for your response.

Response: See response to 1.0Q2.

ii. TS 1.3 - This definition of refueling was deleted. The current TS 1.2 provides a definition of refueling as being the reactor mode when reactivity is $\leq -5 \Delta k/k\%$ and the average reactor coolant temperature is $\leq 140^{\circ}\text{F}$. TS 1.3 states that refueling is "any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted" which is a subset of the mode defined in TS 1.2. The new TS Table 1.1-1 states that refueling is any condition in which "one or more reactor vessel head closure bolt is less than fully tensioned" with fuel in the reactor. While an average reactor coolant temperature or reactivity limit is no longer provided for the refueling mode definition, the reactor vessel head closure bolts cannot be removed at elevated reactor coolant temperatures or when the RCS is pressurized due to their design. A reactivity limit is also not required when the RCS is depressurized. Therefore, the new definition of the refueling mode is more conservative than current TS 1.3 and generally consistent

with TS 1.2. This is a Ginna TS Category (v.a) change.

1.007

The current TS activity and temperature limits have been deleted in the ITS. Why do the ITS operational limits that result from the changes to the refuel definitions result in more restrictive operational requirements for the Ginna Station? Provide supporting justification for your response. What is the minimum temperature and pressure at which the reactor pressure vessel head bolts can be detensioned as allowed by plant procedures? Compare this temperature and pressure to the ITS definition of refueling and explain why there is or is not a change to the existing operational requirements for Ginna? Provide supporting justification for your response.

Response: CTS 1.2 and 1.3 were revised to clarify the head closure status and associated coolant temperatures for plant conditions not previously satisfying a defined MODE or satisfying more than one MODE. This included adding the phrase "all reactor vessel head closure bolts fully tensioned" as a footnote to refueling. The footnote would also apply to the Hot Standby and Cold Shutdown defined modes (i.e., any point at which a reactor head closure bolt was not fully tensioned). Clarifying the shutdown modes with the new footnote eliminates the overlap in defined modes when the temperature is above 140°F and the vessel head is unbolted. It is not the intent of the ITS to allow an option of whether to apply "Refueling" applicable LCOs or to apply "Shutdown" applicable LCOs.

Additionally, the average coolant temperature and associated reactivity condition for the refueling mode are replaced with the notation "NA" consistent with NUREG-1431. The reactivity condition for this mode will be assured by the provisions of ITS LCO 3.9.1. The intent of these changes is to provide clarity and completeness in avoiding any misinterpretation, and as such could be considered administrative. However, since this change eliminates the potential to interpret certain plant conditions such that no MODE, or a less restrictive MODE would exist when above 140°F, this change is discussed and justified as a more restrictive change. With respect to minimum boltup temperatures, this is currently not in TS; however, this value (60°F) will now be in the PTLR.

- iii. TS 1.5 - The definition for Operating was not added to the new specifications since it is no longer required. This definition is addressed by the new definition for OPERABLE - OPERABILITY. This is a Ginna TS Category (i) change.
- iv. TS 1.6 - The definition for Degree of Redundancy (Instrument Channels) was not added to the new specifications since it is no longer required. This definition is addressed within new TS 3.3 (Instrumentation). This is a Ginna TS Category (v.c) change.
- v. TS 1.7.1 - This was revised to specify that the CHANNEL CALIBRATION includes the required interlock and time constant functions of the channel.

1.008

Approval of the use of time constant functions in the definition of

channel calibration requires a WOG traveler for review and approval by the staff. In addition, discussion of calibrating instrument channels with resistance temperature detectors was added for clarification. These are Ginna TS Category (v.a) changes.

Response: RG&E agrees to generate a traveller for this change. Comment #158 has been opened to initiate and track this traveller. [Change later withdrawn by comment #158]

- vi. TS 1.7.2 - The last sentence of this definition was revised as follows:

This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrumentation channels measuring the same parameter.

These minor changes provide greater clarification of the defined term and are Ginna TS Category (v.c) changes.

- vii. TS 1.7.3 - The definitions for testing of analog and bistable channels were combined into one description with a new title. The only difference between the two definitions is that testing of bistable channels required injection of a simulated or source signal into the sensor versus "as close to the sensor as possible" for analog channels.

1.0Q9

Another difference is the use of an "actual" signal in the ITS vice "source signal" in the CTS. Explain why this is or is not a change to the existing operational requirements for Ginna? Provide supporting justification for your response. Since the bistable must be actuated to determine operability, maintaining the analog channel description for the combined definition is acceptable. In addition, the combined definition was expanded to require "adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy." These are Ginna TS Category (v.a) changes.

Response: RG&E agrees that the CTS definitions have also been revised to add the phrase "or actual" in reference to the injected signal as a clarification consistent with NUREG-1431. There is no reason why an actual signal would preclude satisfactory verification or performance of an actuation logic test or channel operational test. Use of an actual signal instead of a simulated signal will not affect the performance of the associated components. Operability can be adequately demonstrated in either case since the associated components cannot discriminate between actual or simulated signals. This is perceived as the intent of the CTS wording, and therefore, the revised wording more accurately reflects this intent and is considered to be administrative.

- viii. TS 1.7.4 - The definition for Source Check was not added to the new specifications since it is no longer required. The performance of a Source Check is now addressed within the definition of CHANNEL CALIBRATION and CHANNEL OPERATING TEST (COT).

1.0Q10 This is a conclusion without an evaluation to support the conclusion. Explain why this is or is not a change to the existing operational requirements for Ginna? Provide supporting justification for your response. This is a Ginna TS Category (v.c) change.

Response: The Source Check definition has been deleted consistent with NUREG-1431. This definition is deleted since the specific CTS referencing the definition is not retained in the ITS. Discussion of the technical aspects of this change is addressed in the Technical Specification where the definition is used. The removal of the definition is considered administrative, with no impact of its own.

ix. TS 1.8 - The definition for Containment Integrity was not added to the new specifications since it is no longer required. Containment Integrity is addressed by new TS 3.6 which essentially requires compliance with 10 CFR 50, Appendix J.

1.0Q11 Explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? Provide supporting justification for your response. This is a Ginna TS Category (v.c) change.

Response: The CTS Containment Integrity definition has been deleted to eliminate the confusion associated with this definition compared to its use in applicable LCOs. The specific requirements denoted in the definition are addressed in ITS LCO 3.6.1 with respect to containment OPERABILITY (versus containment integrity). The deletion of this definition maintains the consistency with NUREG-1431 and is merely an administrative presentation preference.

x. TS 1.10 - The definition for Hot Channel Factors was not added to the new specifications since it is no longer required. The Hot Channel Factor limit is only discussed in one LCO with the limit defined in the COLR.

1.0Q12 Explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? Provide supporting justification for your response. This is a Ginna TS Category (v.c) change.

Response: The CTS Hot Channel Factors definition has been deleted to eliminate the confusion associated with this definition compared to its use in applicable LCOs. The specific requirements denoted in the definition are addressed in ITS Chapter 3.2. Discussion of the technical aspects of the deletion or revision of the applicable TS requirements will be addressed in the TSs where the definition is used (ITS LCO 3.2). The ITS Bases for this LCO also contains a description of what constitutes hot channel factors. The deletion of this definition maintains the consistency with NUREG-1431 and is

merely an administrative presentation preference.

- xi. TS 1.11 - This previously deleted definition was not added to the new specifications. This is a Ginna TS Category (vi) change.
- xii. TS 1.12 - The Frequency for Surveillance Requirements is now specified in hours, days or months in the new specifications such that the current definition of Frequency Notation is no longer required. Consequently, this definition was replaced with a general description of how to use and apply the Frequency requirements. In addition, the definition of refueling Frequency was revised from 18 months to 24 months for all systems. This is discussed in Attachment H and is a Ginna TS Category (v.b.1) change.
- xiii. TS 1.13 - The definition for Offsite Dose Calculation Manual (ODCM) was

1.0Q13

Are the following redline/strikeout text changes appropriate? If so modify the justification, if not provide the appropriate changes to explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? ..

"moved to the ODCM program description in ITS specification 5.5.1. The change to the CTS is editorial because the program description involves the reorganization or reformatting of requirements without affecting technical content"

Response: RG&E agrees to revise the change justification. Comment #157 has been opened to revise the description of change.

This is a Ginna TS Category (v.c) change.

- xiv. TS 1.14 - The definition for Process Control Program (PCP) was not added to the new specifications since it is no longer required. The PCP was relocated from the technical specifications to the TRM and does not need to be described within new TS 1.1. This is a Ginna TS Category (v.c) change.
- xv. TS 1.15 - The definition for Solidification was not added to the new specifications since it is no longer required. Solidification is described within the PCP which was relocated from the technical specifications to the TRM. Therefore, this definition does not need to be provided in new TS 1.1. This is a Ginna TS Category (v.c) change.
- xvi. TS 1.16 - The definition for Purge - Purging was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3.

1.0Q14

Explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? Provide supporting justification for your response. This is a Ginna

TS Category (v.c) change.

Response: The Purge-Purging definition has been deleted consistent with NUREG-1431. This definition is deleted since the specific CTS referencing the definition is not retained in the ITS. Discussion of the technical aspects of this change is addressed in the Technical Specification where the definition is used. The removal of the definition is considered administrative, with no impact of its own.

xvii. TS 1.17 - The definition for Venting was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3.

1.0Q15 Explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? Provide supporting justification for your response. This is a Ginna TS Category (v.c) change.

Response: The Ventilation definition has been deleted consistent with NUREG-1431. This definition is deleted since the specific CTS referencing the definition is not retained in the ITS. Discussion of the technical aspects of this change is addressed in the Technical Specification where the definition is used. The removal of the definition is considered administrative, with no impact of its own.

xviii. TS 1.18 - The reference to the "dose conversion factors for adult thyroid dose via inhalation" was not added to the new specifications since a specific reference to Table E-7 of Regulatory Guide 1.109 was added. This table only contains dose conversion factors for adults via inhalation.

1.0Q16 Provide a CTS markup of this change. Explain how the ITS limits are editorial in nature or involve the reorganization or reformatting of requirements without affecting changes to the existing operational requirements for Ginna? Provide supporting justification for your response. Therefore, the existing reference is no longer necessary. This change is consistent with Traveller WSTS-1, C.2. This is a Ginna TS Category (vi) change.

Response: The definition of DOSE EQUIVALENT I-131 was revised to relocate the details denoting the thyroid dose conversion factors to the bases for ITS LCO 3.4.16 and LCO 3.7.10. This change will make the content of the definition consistent with other definitions and permit future updates of the calculational methods to be revised in accordance with the TS Bases Control Program. This change is consistent with industry proposed Traveller TSTF-03. The requested revised markup of CTS page 1-8 is attached.

xix. TS 1.19 - The definition for Reportable Event was not added to the new specifications since it is no longer required. Reportable Events are described in 10 CFR 50.72 and 50.73. This is a Ginna TS

Category (ii) change.

- xx. TS 1.20 - The definition for Canisters Containing Consolidated Fuel Rods was not added to the new specifications since it is no longer required. This definition is moved to new TS 4.3 which is the only section that addresses consolidated fuel rods. This is a Ginna TS Category (v.c) change.
- xxi. TS 1.21 - The definition for Shutdown Margin was expanded to require another assumption that in MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature. Also, the definition was revised to require consideration of any RCCA known to be incapable of being fully inserted. This is in addition to the existing assumptions related to a stuck fully withdrawn single RCCA with the highest reactivity worth. The definition description discussing "no changes in xenon or boron concentration" was deleted since this level of detail is not required. These clarifications, which are consistent with NUREG-1431, are Ginna TS Category (v.a) changes.
- xxii. TS 1.4 - The definition for OPERABLE - OPERABILITY was revised to remove "supports." This phrase was added to the current definition by Reference 3 but is not consistent with the definition as provided in NUREG-1431. Therefore, to provided consistency, this was not added to the new specifications. This is a Ginna TS Category (v.c) change.
- xxiii. The following definitions were added to the new specifications since the associated terms are used throughout the document (these are Ginna TS Category (v.a) changes):
 - a. ACTIONS
 - b. ACTUATION LOGIC TEST
 - c. AXIAL FLUX DIFFERENCE
 - d. CORE ALTERATION
 - e. CORE OPERATING LIMITS REPORT (COLR)
 - f. LEAKAGE
 - g. PHYSICS TESTS
 - h. PRESSURE TEMPERATURE LIMITS REPORT (PTLR)
 - i. RATED THERMAL POWER
 - j. STAGGERED TEST BASIS
 - k. TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)
- xxiv. A new section was added to the specifications which explains the use of Logical Connectors within the new TS. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.
- xxv. A new section was added to the specifications which explains the use of the Completion Time convention within the new TS. There are several changes from the current Ginna Station TS format which are discussed in this section (these are Ginna TS Category (v.a) changes):



- a. Completion Times in the new TS are based on the format that the clock for all Required Actions begin from the time that the Condition is entered. The Completion Times in the new specifications and the current Ginna Station TS are typically equal. For example, the new specifications may require that the plant be in MODE 3 within 6 hours and in MODE 4 within 36 hours for a specified Condition while the current Ginna Station TS require that the plant be in MODE 3 within 6 hours and in MODE 4 within an additional 30 hours for the same Condition. The intent of both the new specifications and the current Ginna Station TS is the same (i.e. be in MODE 4 within 36 hours).
 - b. The new specifications restrict multiple entries into the ACTION table for separate Conditions unless it is specifically stated as acceptable. For example, if one SI pump is inoperable and during the LCO, a second SI pump is declared inoperable, the plant would enter 3.0 conditions in both the new specifications and the current Ginna Station TS. If the first SI pump were restored to OPERABLE status before entering MODE 3, the plant could resume operation in both TS. However, in the current TS, the Completion Time for restoring the second SI pump to OPERABLE status would begin from the time that it was declared inoperable. In the new specifications, the Completion Time would begin from the time the first pump was declared inoperable with an additional 24 hours allowed. This is a conservative change.
- xxvi. A new section was added to the specifications which explains the use of the Frequencies specified within the SRs. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.

Section 1.0 Improved TS

1.ITS 1.1

- i. Incorporation of approved Traveller WOG-01, C.1.
- ii. Incorporation of approved Traveller BWOG-01, C.1.
- iii. Incorporation of approved Traveller BWR-05, C.1. This traveller was also revised to replace the second use of "cross calibration" in the CHANNEL CALIBRATION definition with "qualitative assessment of sensor behavior."

1.0Q17

Confirm the revised traveler changes, and if necessary correct the markup of the Channel Calibration definition to match the approved traveler. This change provides consistency within the definition

and is an ITS Category (iii) change.

Response: The definition of CHANNEL CALIBRATION was revised beyond the changes identified in BWR-05, C.1 to replace "cross calibration" with "qualitative assessment of sensor behavior." The change is similar to the change approved in BWR-05, C.1. The term "cross calibration" implies activities which are not possible on RTDs or thermocouples. "Calibrations" typically require adjustments of devices to cause them to conform to a desired output. In this sense, RTDs and thermocouples cannot be calibrated. The appropriate activity to require on an RTD or thermocouple is to cross compare RTD or thermocouple output indications from sensors measuring the same temperature. Therefore, this proposed change is intended to provide a more appropriate presentation of the intended requirement.

- iv. Incorporation of approved Traveller BWR-05, C.3, and approved Traveller BWO-01, C.3.
- v. Minor changes were made to the Definitions and the Completion Time and Frequency discussions to provide consistency within the new specifications and bases. Examples include the use of "plant" versus "unit" since there is only one nuclear unit at Ginna Station,

1.0Q18 Replacing "unit" with "plant" is acceptable

Response: No response required.

1.0Q19 Specifying that the LEAKAGE definition is related to the RCS is not acceptable because not all terms used in the definition refer to RCS leakage.

Specifying that the LEAKAGE definition is related to the RCS

Response: All uses of LEAKAGE in the ITS and in this definition refer to RCS leakage. The addition of this text was requested by Ginna operations since CTS 3.1.5.2.1 specifically uses "RCS leakage" for this definition. While it could be argued that unidentified LEAKAGE could be interpreted as from unknown sources other than RCS, the addition of "RCS" is provided to clarify this misinterpretation since this is meant to be unidentified leakage from the RCS that has not been placed in the identified LEAKAGE bin or determined to be from a RCS pressure boundary. This interpretation is also consistent with the bases for ITS LCO 3.4.13 which uses this definition. Therefore, RG&E believes the change is necessary and correct.

and editing the AVERAGE DISINTEGRATION ENERGY definition for better readability

1.0Q20 Stet the proposed deletion of "(in MeV)". This is the energy associated with beta and gamma releases on a per disintegration basis. Otherwise provide an NEI traveler for this change. These are ITS Category (iv) changes.

Response: The words "(in MeV)" were proposed to be moved, not deleted, to editorially enhance the definition of Ebar at the request of Ginna radiation protection personnel. The phrase is proposed to be revised to "...beta and gamma energies (in MeV) per disintegration..." This change was rejected as a technical change by the WOG, therefore no traveller has been generated.

vi. The definitions for ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, REACTOR TRIP SYSTEM (RTS) RESPONSE TIME, MASTER RELAY TEST, and SLAVE RELAY TEST, were not added to the new specifications. The current Ginna Station TS do not require ESF or RTS response time testing, nor master and slave relay testing. These requirements are not being added to the new specifications consistent with Reference 2. Therefore, these definitions are not applicable. These are ITS Category (i) changes.

vii. Not used.

viii. Incorporation of approved Traveller BWOG-09, C.26.

ix. Incorporation of approved Traveller BWR-02, C.4.

x. The definition of QUADRANT POWER TILT RATIO (QPTR) was replaced with the definition provided in current Ginna Station TS 1.9. The use of the ITS definition for QPTR would require modifications to the Ginna Station process computer, procedures, and operator training. The current QPTR definition was added to the Ginna Station TS by References 4 and 5.

1.0Q21 Explain why the proposed ITS definition, without the last sentence of the CTS definition, is or is not a change to the existing operational requirements for Ginna. Provide supporting justification for your response. This is an ITS Category (i) change.

Response: The last sentence of CTS 1.9 has been relocated to Notes 2 and 3 for ITS SR 3.2.4.2. These notes state that if $< 75\%$ RTP with one power range channel inoperable, then use the remaining three channels to verify QPTR. If $\geq 75\%$ RTP and more than one power range channel inoperable, ITS SR 3.2.1.2 and SR 3.2.2.2 must be performed. These two SRs require a core flux map to confirm that QPTR is within limits which goes beyond CTS requirements. Consequently, the ITS requirements are actually more conservative than CTS.

xi. Incorporation of approved Traveller BWR-18, C.2.

xii. Incorporation of approved Traveller BWOG-01, C.4.

1.0Q22 Justify the changes to the [] statement in the ISTS definition of shutdown margin

Response: The ITS bracketed phrase "nominal zero power design level" was revised to "nominal hot zero power temperature" to more accurately reflect plant specific requirements. This change is considered an editorial enhancement consistent with plant nomenclature and design

and the use of "hot zero power" throughout the ITS and NUREG (e.g., see bases for LCO 3.4.2).

- xiii. The titles for MODES 3 and 4 were switched. This change provides consistency with the current Ginna Station TS, and the nomenclature used in procedures, the UFSAR and other documents. The revision of all of these documents and operator training materials would require significant resources without any benefit. This is an ITS Category (i) change.

1.0Q23 Provide documentation that the [] average coolant temperatures for the specified modes is consistent with Ginna CTS procedures.

Response: Copies of startup procedures have been provided (attached). See also response to 1.0Q2.

- xiv. The definition of CHANNEL CALIBRATION, COT and TADOT was revised to delete the display requirement on the basis that it will create confusion with respect to establishing the OPERABILITY of a channel. These changes are consistent with Traveller WSTS-1, C.1. This is an ITS Category (iii) change.

- xv. The definition of DOSE EQUIVALENT I-131 was revised to delete and relocate to the Bases for LCO 3.4.16 and LCO 3.7.10 the details denoting the thyroid dose conversion factors. This allows future updates of the calculational methods to be revised without having to change the Technical Specifications. As a result of the proposed relocation of information, the approved Traveller BWOG-01, C.2, was not incorporated. This change is consistent with Traveller WSTS-1, C.2. This is an ITS Category (iv) change.

1.0Q24 These generic changes require submittal of an NEI traveler.

Response: This change has been submitted as an industry proposed traveller TSTF-03. [This change was later withdrawn based on NRC rejection of traveller. See comment #233.]

- xvi. The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) was revised by deleting the references to the LCO's to be consistent with the definition of the COLR and to provide a complete description of its content. This change is consistent with Traveller WSTS-1, C.3. This is an ITS Category (iv) change.

1.0Q25 These generic changes require submittal of an NEI traveler.

Response: This change has been submitted as an industry proposed traveller TSTF-04.

- xvii. Incorporation of approved Traveller WOG-35, C.1.

- xviii. Incorporation of approved Traveller BWR-14, C.1.

- xix. The CHANNEL CALIBRATION definition was revised to include "time constants". This change enables Notes associated with time constants to be removed from Chapter 3.3. This is an ITS Category

(i) change.

2. ITS 1.2

- i. Incorporation of approved Traveller BWOG-01, C.5.
- ii. The term "surveillances" is deleted since logical connectors are not used with respect to surveillances in NUREG-1431. This is an ITS Category (iv) change.
- iii. Appropriate titles were applied to each example. This is an ITS Category (iv) change.

1.0Q26 Editorial comment. Recommended markup for page 1.3-3; 1.3-4 of the markup

Response: This issue will be discussed during the meeting.

3. ITS 1.3

- i. Incorporation of approved Traveller BWR-02, C.5.
- ii. Incorporation of approved Traveller BWOG-01, C.7.
- iii. Incorporation of approved Traveller BWR-02, C.7.
- iv. Incorporation of approved Traveller BWOG-01, C.8.
- v. Incorporation of approved Traveller BWR-06, C.3 (Rev. 1).
- vi. Incorporation of approved Traveller WOG-32, C.1.
- vii. Minor changes to Example 1.3-2 and Example 1.3-6 were made to provide additional clarification. These changes do not alter the intent of the examples. This is an ITS Category (iv) change.

1.0Q27 Editorial comment: o.k., except for the following: Changing "one" to "either." the choice of wording in the ISTS is consistent with the discuss in the second paragraph on page 1.3-4 of the markup; and adding "until the LCO is met" wording to the last paragraph of the ISTS on page 1.3-11 of the markup

Response: RG&E agrees to revise as requested. Comment #159 has been opened to address these items.

- viii. Incorporation of approved Traveller BWOG-01, C.9.
- ix. The Completion Time description was revised to eliminate confusion regarding the applicability of the Required Actions given additional failures in the absence of provisions for separate condition entry. This is an ITS Category (iv) change.

1.0Q28 These generic changes require submittal of an NEI traveler.

Response: This change was proposed for clarity only and is considered an editorial enhancement. For this reason, the revision was rejected as a technical change by the WOG and no traveller has been generated. Comment #160 has been opened to remove this clarification since RG&E no longer views it as "critical" for understanding the affected section.

- x. An additional statement was added to reinforce that, in the application of Completion Time extensions, no single component, subsystem, or variable, etc., can be allowed to remain inoperable for longer than the stated Completion Time. This is an ITS Category (iv) change.

1.0Q29 Editorial comment. Insert the completion time discussion after the first sentence at the top of page 1.3-2 of the markup

Response: RG&E agrees to revise the ITS as requested. Comment # 161 has been opened to revise this clarification.

- xi. This section refers to Completion Times on a "once per" basis, but no example is referenced. An appropriate example was added. This is an ITS Category (iv) change.
- xii. The Completion Time logical connector for Example 1.3-3 was deleted since this connector is not used as discussed in the changes for applicable LCOs. This is an ITS Category (iii) change.

4. ITS 1.4

- i. Incorporation of approved Traveller BWR-05, C.14.
- ii. Minor changes were made to Example 1.4-1 to eliminate redundant text. These changes do not alter the intent of the example. This is an ITS Category (iv) change.

1.0Q30 Editorial comment. Let's discuss the changes proposed to page 1.4-2 of the markup, some valuable discussion useful to understanding how to apply SR 3.0.1 have been deleted.

Response: RG&E agrees to discuss this during the upcoming meetings. Essentially, the text was proposed to be deleted since it reiterated the last paragraph on the same page. [Change was later withdrawn based on meeting week of 11/20/95. See comment #228.]

5. ITS 2.1.1

- i. SL 2.1.1 was revised to delete the reference to the highest loop average. This is base on Ginna Station designed with two loops. The RCS average temperature trips the reactor on coincidence two-out-of-four signals, with two channels per loop. This is an ITS Category (iv) change.
- ii. ITS Figure 2.1.1-1 was replaced with the existing Technical Specification Figure 2.1-1. The Reactor Core Safety Limits (SLs) figure reflects the acceptable operating regions of the plant and is

consistent with the current safety analysis. This is an ITS Category (iv) change.

iii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Ginna Station was designed and built prior to the issuance of the GDC contained in 10 CFR 50, Appendix A. However, the draft GDC issued by the Atomic Industrial Forum (AIF) in 1967 were utilized in the design of Ginna Station. The bases were revised to reflect this difference.
- b. The discussion of DNB criteria was revised and expanded to reflect plant-specific considerations.
- c. Various wording changes were made to improve the readability and understanding of the bases and to reflect plant-specific considerations.
- d. The listing of the automatic functions relating to the enforcement of the reactor core SLs was revised consistent with the changes proposed in ITS Chapter 3.3.
- e. Ginna Station was analyzed for the locked rotor event to show that the peak reactor coolant system pressure remains below 120% of design. The bases were revised to reflect this difference.
- f. The additional words to the reference allow for approved exceptions.
- g. A typographical or minor clarification is identified.

iv. Incorporation of approved Traveller WOG-01, C.1.

v. Incorporation of approved Traveller BWR-11, C.10 (Rev.1).

6. ITS 2.2

i. SL 2.2.3 - This section and associated bases were not added. The section duplicates a reporting requirement described in 10 CFR 50.36(c)(1) and 10 CFR 50.72. The deletion of this section eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents. This change is consistent with Traveller WSTS-2, C.1. This is an ITS Category (i) change.

ii. SL 2.2.4 - This section and associated bases were not added. This requirement for the notification of management personnel and plant safety review committees is similar to the requirements removed from other sections of the ITS (i.e., Chapter 5.0 - "Administrative

Controls" for the Onsite and Offsite review function) and relocated to other licensee controlled documents. The relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety. This change is consistent with Traveller WSTS-2, C.1. This is an ITS Category (i) change. As a result of this TS change, approved Traveller WOG-21, C.1 and C.2 were not incorporated.

- iii. SL 2.2.5 - This section and associated bases were not added. The section, in part, duplicates a reporting requirement described in 10 CFR 50.36(c)(1) and 10 CFR 50.73. The deletion of this requirement eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents. The section also requires distribution of the safety violation report to certain management personnel and plant safety review committees. This requirement is similar to the requirements removed from other sections of the ITS (i.e., Chapter 5.0 - "Administrative Controls" for the Onsite and Offsite review function) and relocated to other licensee controlled documents. The relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety. This change is consistent with Traveller WSTS-2, C.1. As a result of this change, approved Travellers WOG-21, C.1 and C.2, and BWR-02, C.8 and C.8a were not incorporated. This is an ITS Category (i) change.
 - iv. SL 2.2.6 - This section and associated bases were not added. The section duplicates a requirement described in 10 CFR 50.36(c)(1). The deletion of this section eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents. This change is consistent with Traveller WSTS-2, C.1. This is an ITS Category (i) change.
 - v. Various wording changes were made to improve the readability and understanding of the bases and to reflect plant-specific considerations. This is an ITS Category (iv) change.
7. ITS 3.0
- i. For completeness LCO 3.0.7 should also be referenced in LCO 3.0.1. As discussed in approved Traveller NRC-03, C.5, LCO 3.0.7 addresses a situation when an LCO requirement is allowed not to be met. This is similar to LCO 3.0.2 which addresses the requirement of meeting

the associated ACTIONS when not meeting a Limiting Condition for Operation. This change is consistent with Traveller WSTS-3, C.1. This is an ITS Category (iii) change.

- ii. Incorporation of approved Traveller BWOG-01, C.10.
- iii. Incorporation of approved Traveller BWR-02, C.11.
- iv. Incorporation of approved Traveller BWR-05, C.7.
- v. Incorporation of approved Traveller BWOG-09, C.26 (Rev 1).
- vi. Incorporation of approved Traveller NRC-03, C.5 (Rev 1). Minor changes are made to reflect the actual proposed new TS. These are ITS Category (iv) changes.
- vii. Incorporation of approved Traveller BWR-02, C.10.
- viii. Incorporation of approved Traveller BWR-25, C.3.
- ix. Incorporation of approved Traveller BWR-05, C.10 (Rev 2). Several wording changes were made to increase understanding. These changes do not alter the indent of the Traveller. These are ITS Category (iv) changes.
- x. Incorporation of approved Traveller BWR-05, C.12.
- xi. Incorporation of approved Traveller BWR-05, C.15 and WOG-01, C.2.
- xii. Incorporation of approved Traveller BWR-05, C.13 (Rev 2).
- xiii. A typographical or minor clarification is identified. This is an ITS Category (iv) change.
- xiv. Incorporation of approved Traveller BWOG-01, C.11.
- xv. Incorporation of approved Traveller BWR-07, C.1 (Rev 1).
- xvi. LCO 3.0.3 and the bases were revised to remove the requirement to initiate action to shutdown the plant within 1 hour. Instead, the bases require the Shift Supervisor to evaluate the plant conditions to determine if a plant shutdown should be initiated immediately, or deferred if the condition which caused entry into LCO 3.0.3 is expected to be restored within a reasonable period of time. However, the time restrictions in LCO 3.0.3 for MODE changes must always be met. This change provides the plant management and operating staff with the flexibility to determine the best course of action should LCO 3.0.3 be entered. This change is consistent with Traveller WSTS-3, C.2. This is an ITS Category (i) change.
- xvii. The bases for SR 3.0.1 were revised to clarify that credit may be taken for unplanned events that satisfy the performance of an SR. This change allows the deletion of multiple Notes within the SRs in Chapter 3 which state the same thing. The change is consistent with Traveller WSTS-3, C.3. This is an ITS Category (iii) change.

- xviii. LCO 3.0.4 and the bases were revised to provide greater clarity and consistency with actual Ginna Station practices. First, the details of why exceptions are allowed to LCO 3.0.4 was deleted from the LCO and relocated to the bases. This change provides consistency with LCO 3.0.3 and SR 3.0.2. Second, the bases were revised to provide easier readability. In addition, current Ginna Station operating practices prevent any MODE change, up or down, with inoperable equipment required for the MODE desired to be entered. Therefore, the discussion that LCO 3.0.4 does not prevent MODE changes during a "normal shutdown" conflicts with these practices and was deleted. These are ITS Category (iii) changes.

Section 2.0 & 3.0 Current TS

2. Technical Specification 2.1

- i. The Applicability was revised to not only include when the reactor is in "operation" or critical, but also when in MODE 2 and subcritical. This ensures that the Reactor Core Safety Limits are also met during reactor startup since there is a potential for an inadvertent criticality with the reactor near normal operating temperature and pressure conditions. This is a Ginna TS Category (iv.a) change.

2.0Q1 The existing TS 2.1 reactor core safety limit applicability was stated to be revised in improved TS 2.1.1 to not only include when the reactor is in "operation" or critical, but also when in MODE 2 and subcritical. ITS Mode 2 is critical operation. Clarify justification 2.i.

Response: RG&E agrees to revise the CTS markup and this change justification as follows:

- i. *The Applicability was revised to define when the reactor is in "operation" as MODES 1 and 2. This is an editorial change only since "operation" has been redefined as MODES 1 and 2 per change D.1.i.d. This is a Ginna TS Category (iv.a) change.*

2.0Q2 The safety limit specification references on page 2.1-3 list the FSAR and a NRR staff safety evaluation for the 95/95 fuel cladding damage margin. These references are deleted. Justify the change. Why did references change to UFSAR from FSAR?

Response: These references are in the Bases section of the CTS and are not specifically required to be justified since the CTS bases are being replaced in their entirety. The corresponding ITS Bases adequately describes this level of detail. The ITS Bases reflects references from the "UFSAR" rather than the "FSAR" since the FSAR was superseded by the UFSAR in 1984. The CTS bases were not revised at that time to reflect the use of the UFSAR since it was assumed to require a TS amendment. Instead, a cross-matrix was generated internally to address this issue. It should also be noted that Westinghouse has reviewed and approved the bases proposed in the ITS Chapter 2.0 at the request of the Ginna Station PORC.

3. Technical Specification 2.2

- i. The Applicability was revised to "MODES 1, 2, 3, 4, and 5." The proposed Applicability does not require this Safety Limit (SL) to be met when fuel is in the vessel with one or more reactor vessel head closure bolts less than fully tensioned or with the head removed. With the reactor head bolts less than fully tensioned, it is highly unlikely that the RCS can be pressurized greater than the SL pressure due to the low temperature over-pressure protection requirements. With the head removed, it is not possible to pressurize the RCS greater than the SL pressure. This is a Ginna TS Category (v.b.2) change.

2.0Q3 The safety limit specification references on page 2.2-2 list the FSAR for the RCS 110% limit of 2735 psig. These references are deleted. Justify the change. Why did references change to UFSAR from FSAR?

Response: See response to 2.0Q2.

4. Technical Specification 2.3

- i. This entire section was relocated to ITS Chapter 3.3, "Instrumentation." This is a Ginna TS Category (i) change.
- ii. TS 2.3 - Various limiting safety system settings (LSSS) are addressed as "Trip Setpoints," "Allowable Values," or "Applicable Modes" (as permissives) for their respective Reactor Trip System (RTS) instrumentation Functions in new LCO 3.3.1. Specific changes to the LSSS are discussed below for each of the associated Functional Units. This is a Ginna TS Category (i) change.
- iii. TS 2.3.1.2.d and TS 2.3.1.2.e - Various parameters used in the methodology for determining the Overtemperature ΔT and the Overpower ΔT Functions were not added to the specifications. These parameters are associated with variables which may change as a result of a reload analysis and are relocated to the COLR. This is a Ginna TS Category (iii) change.
- iv. TS 2.3.3.1, TS 2.3.3.2, and Figure 2.3-1 - The LSSS for the loss of voltage and degraded voltage functions were revised to provide a minimum Trip Setpoint value. Criteria for the establishment of equivalent values based on measured voltage versus relay operating time was relocated to the bases for LCO 3.3.4. This is a Ginna TS Category (iii) change.
- v. TS 2.3.1.2.g - The LSSS for the RCP underfrequency Functions was not added to the new specifications. This is justified in Reference 44 which shows that this trip function, though installed at Ginna Station, is not required or applicable based on the offsite power source design. This setpoint and requirement are relocated to the TRM. This is a Ginna TS Category (iii) change.

55. Technical Specification 6.7

- i. TS 6.7.1.a - The initial operator actions for Safety Limit (SL) violations were revised as follows:
 - a. For violation of the Reactor Core or RCS Pressure SL in MODES 1 and 2, the requirement to immediately shutdown the reactor (effectively to be in MODE 3) was revised to allow 1 hour to restore compliance and place the unit in MODE 3. Immediately shutting down the reactor could infer action to immediately trip the reactor. The revision provides the necessary time to shutdown the unit in a more controlled and orderly manner than immediately tripping the reactor which could result in a plant transient. The proposed time continues to minimize the time allowed to operate in MODE 1 or 2 with a SL not met. This is a Ginna TS Category (v.b.44) change.
 - b. For violation of the RCS Pressure SL in MODES 3, 4, and 5, an additional action was added which requires restoring compliance with the SL within 5 minutes. Specifying a time limit for operators to restore compliance provides greater guidance to plant staff. This is a Ginna TS Category (v.a) change.
- ii. TS 6.7.1.b - The requirement for notification to management personnel and the offsite review function of a SL violation was not added to the new specifications. Notification requirements are relocated to the TRM. This is a Ginna TS Category (iii) change. The requirement for notification to the NRC of a SL violation was not added to the new specifications since this requirement is denoted in 10 CFR 50.36 and 10 CFR 50.72. This is a Ginna TS Category (ii) change.

2.0Q4 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

- iii. TS 6.7.1.c - The requirement that a Safety Limit Violation Report be prepared was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the onsite review committee to review the Safety Limit Violation Report was not added to the new specifications. The responsibilities of the onsite review committee are relocated to the TRM. This is a Ginna TS Category (iii) change. SL violations are reported to the NRC in accordance with the provisions of 10 CFR 50.73. The details describing the requirements for content of the Safety Limit Violation Report is, therefore, controlled by the provisions of 10 CFR 50.73 and does not need to be specified in TS. This is a Ginna TS Category (ii) change.

2.0Q5 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

- iv. TS 6.7.1.d - The requirement for the submittal of a Safety Limit Violation Report to the NRC was not added to the new specifications.

This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the submittal of a Safety Limit Violation Report to management personnel and the offsite review function was not added to the new specifications. The distribution of reports submitted in accordance with 10 CFR 50.73 are relocated to the TRM. This is a Ginna TS Category (iii) change.

2.0Q6 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

5. Technical Specification 3.0

- i. A new section LCO 3.0.1 was added which explains the use of the Applicability statement in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.1. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- ii. A new section LCO 3.0.2 was added which explains the use of the associated ACTIONS upon discovery of a failure to meet an LCO in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.2. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- iii. TS 3.0.1 - This was revised to clarify the use of the actions that must be implemented when an LCO is not met and (1) an associated Required Action and Completion Time is not met and no other Condition applies, or (2) the condition of the plant is not specifically addressed by the associated ACTIONS. The current requirement that the LCO time limits apply if they are more limiting than those required by LCO 3.0.3 is deleted and an expanded discussion is provided in the Basis to clarify the applicability of this requirement. This section does not provide any new requirements. The clarifications and examples are based on the use of the new ITS format. This is a Ginna TS Category (v.c) change.
- iv. A new section LCO 3.0.4 was added which explains the limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met in the new TS. This section provides new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.
- v. A new section LCO 3.0.5 was added to provide an exception to LCO 3.0.2 for instances where restoration of inoperable equipment to an OPERABLE status could not be performed while continuing to comply

with Required Actions. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as: maintaining an isolation valve closed or tripping an inoperable instrument channel. To allow the performance of SRs to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the present Specifications. Without this allowance certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications to preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. Since this requirement is informally utilized and has no licensing basis, this section is considered to provide new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.

- vi. TS 3.0.2 - This was deleted and replaced by LCO 3.0.6 which provides guidance regarding the appropriate ACTIONS to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). Since its function is to clarify existing ambiguities and to maintain actions within the realm of previous industry interpretations and NRC positions, this new provision does not provide any new requirements. The information contained in TS 3.0.2 was relocated to LCO 3.8.1 which allows one power source to a safeguards bus and a redundant safety features on a second bus to be inoperable for 12 hours versus 1 hour. This change is consistent with NUREG-1431. These are Ginna TS Category (v.c) and (i) changes, respectively.
- vii. A new section LCO 3.0.7 was added to provide guidance regarding Test Exceptions for LCO 3.1.8. This LCO allows specified Technical Specification requirements to be changed (i.e., made applicable in part or whole, or suspended) to permit the performance of special tests or operations which otherwise could not be performed. If this Test Exception LCO did not exist, many of the special tests and operations necessary to demonstrate select plant performance characteristics, special maintenance activities and special evolutions could not be performed. This Specification eliminates the confusion which would otherwise exist as to which LCOs apply during the performance of a special test or operation. Without this specific allowance to change the requirements of another LCO, a conflict of requirements could be incorrectly interpreted to exist. This section does not provide any new requirements. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.

27. Technical Specification 4.0

- i. A new section SR 3.0.1 was added which establishes the requirements and limitations that the SRs must meet during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply.
- ii. TS 4.0 was revised to clarify the basic application of the 25% extension to routine surveillances consistent with the use and format of the ITS.
- iii. A new section SR 3.0.3 was added which establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency.
- iv. A new section SR 3.0.4 was added which establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

Section 2.0 Improved TS

5. ITS 2.1.1

- i. SL 2.1.1 was revised to delete the reference to the highest loop average. This is based on Ginna Station designed with two loops. The RCS average temperature trips the reactor on coincidence two-out-of-four signals, with two channels per loop. This is an ITS Category (iv) change.
- ii. ITS Figure 2.1.1-1 was replaced with the existing Technical Specification Figure 2.1-1. The Reactor Core Safety Limits (SLs) figure reflects the acceptable operating regions of the plant and is consistent with the current safety analysis. This is an ITS Category (iv) change.
- iii. The bases were revised as follows (these are ITS Category (iv) changes):
 - a. Ginna Station was designed and built prior to the issuance of the GDC contained in 10 CFR 50, Appendix A. However, the draft GDC issued by the Atomic Industrial Forum (AIF) in 1967 were utilized in the design of Ginna Station. The bases were revised to reflect this difference.
 - b. The discussion of DNB criteria was revised and expanded to reflect plant-specific considerations.
 - c. Various wording changes were made to improve the readability and understanding of the bases and to reflect plant-specific considerations.

2.007

Bases insert 2.1.3 for page B 2.0-3 is incomplete; provide a discussion of the second slope of figure 2.1.1-1 and correct the awkward wording of the last sentence, "such that overtemperature the

hot leg steam quality....".

Response: Insert 2.1.3 has been revised consistent with the wording provided in Attachment C to the submittal (attached). This wording was revised at the last second by PORC without a corresponding rewording of the insert in the NUREG markup.

2.0Q8 Status: reject Bases B 2.0-8 proposed deletion of safety limit USAR reference.

Response: All references on NUREG B 2.0-8 has only been updated (i.e., it is the same reference source, only the reference number has been changed due to the order in which the reference is called upon by the bases). The only exception is the deleted "(Ref. 4)" in the first line of the second paragraph. This deletion was made since Reference 4, or 10 CFR Part 100, is not the bases for Reactor Trip System setpoints.

d. The listing of the automatic functions relating to the enforcement of the reactor core SLs was revised consistent with the changes proposed in ITS Chapter 3.3.

2.0Q9 Explain why the pressurizer trip on high and low pressure are deleted and not otherwise discussed in the Bases. The NUREG markup includes these trips as function 7. Also justify deleting the enthalpy discussion.

Response: These functions were deleted because they fall into the category of trip functions which are provided to backup the primary Functions (e.g., those identified in the list) for specific abnormal conditions. This backup category has been included as a statement following the list of functions. Please note that these bases changes and clarifications were provided by Westinghouse during the preparation of this section. Essentially, the four items listed in the Applicable Safety Analyses bases are the only items specifically credited in the accident analyses for Westinghouse plants:

With respect to the enthalpy discussion, this has been relocated to the Safety Limits section (i.e., Insert 2.1.3) where it is more relevant and easier to understand. [This response was revised as a result of meetings the week of 11/1/95. See comment #219.]

e. Ginna Station was analyzed for the locked rotor event to show that the peak reactor coolant system pressure remains below 120% of design. The bases were revised to reflect this difference.

f. The additional words to the reference allow for approved exceptions.

g. A typographical or minor clarification is identified.

iv Incorporation of approved Traveller WOG-01, C.1.

v. Incorporation of approved Traveller BWR-11, C.10 (Rev.1).

6. ITS 2.2

- i. SL 2.2.3 - This section and associated bases were not added. The section duplicates a reporting requirement described in 10 CFR 50.36(c)(1) and 10 CFR 50.72. The deletion of this section eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents. This change is consistent with Traveller WSTS-2, C.1. This is an ITS Category (i) change.

2.0Q10 These generic changes require a staff approved traveller.

Response: The change has been submitted as an industry proposed traveller TSTF-05.

- ii. SL 2.2.4 - This section and associated bases were not added. This requirement for the notification of management personnel and plant safety review committees is similar to the requirements removed from other sections of the ITS (i.e., Chapter 5.0 - "Administrative Controls" for the Onsite and Offsite review function) and relocated to other licensee controlled documents. The relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety. This change is consistent with Traveller WSTS-2, C.1.

2.0Q11 These generic changes require a staff approved traveller. This is an ITS Category (i) change. As a result of this TS change, approved Traveller WOG-21, C.1 and C.2 were not incorporated.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

- iii. SL 2.2.5 - This section and associated bases were not added. The section, in part, duplicates a reporting requirement described in 10 CFR 50.36(c)(1) and 10 CFR 50.73. The deletion of this requirement eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents. The section also requires distribution of the safety violation report to certain management personnel and plant safety review committees. This requirement is similar to the requirements removed from other sections of the ITS (i.e., Chapter 5.0 - "Administrative Controls" for the Onsite and Offsite review function) and relocated to other licensee controlled documents. The

relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety.

2.0Q12 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

This change is consistent with Traveller WSTS-2, C.1. As a result of this change, approved Travellers WOG-21, C.1 and C.2, and BWR-02, C.8 and C.8a were not incorporated. This is an ITS Category (i) change.

- iv. SL 2.2.6 - This section and associated bases were not added. The section duplicates a requirement described in 10 CFR 50.36(c)(1). The deletion of this section eliminates the need to change technical specifications when there are rule changes. Since RG&E must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicate requirements from technical specifications. The implementation of these requirements are contained in procedures or other controlled licensee controlled documents.

2.0Q13 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-05.

This change is consistent with Traveller WSTS-2, C.1. This is an ITS Category (i) change.

- v. Various wording changes were made to improve the readability and understanding of the bases and to reflect plant-specific considerations. This is an ITS Category (iv) change.

Section 3.0 Improved TS

7. ITS 3.0

- i. For completeness LCO 3.0.7 should also be referenced in LCO 3.0.1. As discussed in approved Traveller NRC-03, C.5, LCO 3.0.7 addresses a situation when an LCO requirement is allowed not to be met. This is similar to LCO 3.0.2 which addresses the requirement of meeting the associated ACTIONS when not meeting a Limiting Condition for Operation. This change is consistent with Traveller WSTS-3, C.1. This is an ITS Category (iii) change.

3.0Q1 These generic changes require a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-06.

- ii. Incorporation of approved Traveller BWOG-01, C.10.
- iii. Incorporation of approved Traveller BWR-02, C.11.
- iv. Incorporation of approved Traveller BWR-05, C.7.
- v. Incorporation of approved Traveller BWOG-09, C.26 (Rev 1).
- vi. Incorporation of approved Traveller NRC-03, C.5 (Rev 1). Minor changes are made to reflect the actual proposed new TS. These are ITS Category (iv) changes.
- vii. Incorporation of approved Traveller BWR-02, C.10.
- viii. Incorporation of approved Traveller BWR-25, C.3.
- ix. Incorporation of approved Traveller BWR-05, C.10 (Rev 2). Several wording changes were made to increase understanding. These changes do not alter the indent of the Traveller. These are ITS Category (iv) changes.
- x. Incorporation of approved Traveller BWR-05, C.12.
- xi. Incorporation of approved Traveller BWR-05, C.15 and WOG-01, C.2.
- xii. Incorporation of approved Traveller BWR-05, C.13 (Rev 2).
- xiii. A typographical or minor clarification is identified. This is an ITS Category (iv) change.

3.0Q2 These proposed change to Bases page B 3.0-10 is generic and requires a staff approved NEI traveller.

Response: This was provided for clarity and is considered an editorial enhancement. For this reason, the revision was rejected as a technical change by the WOG and no traveller has been generated. Comment #162 has been generated to delete this inserted text.

3.0Q3 The proposed change to LCO 3.0.3 Bases on page B 3.0-4 is generic and requires a staff approved NEI traveller.

Response: The proposed change (other than replacing "unit" with "plant" and modifying LCO numbers and titles) modifies the first two sentences of the second paragraph and is considered an editorial enhancement. For this reason, the revision was rejected as a technical change by the WOG and no traveller has been issued. These two sentences in the NUREG imply that there are different time limits specified in LCO 3.0.3 depending on which MODE the plant is originally in. This is not the case as provided in the discussion and examples which follow this text.

- xiv. Incorporation of approved Traveller BWOG-01, C.11.
- xv. Incorporation of approved Traveller BWR-07, C.1 (Rev 1).

xvi. LCO 3.0.3 and the bases were revised to remove the requirement to initiate action to shutdown the plant within 1 hour. Instead, the bases require the Shift Supervisor to evaluate the plant conditions to determine if a plant shutdown should be initiated immediately, or deferred if the condition which caused entry into LCO 3.0.3 is expected to be restored within a reasonable period of time. However, the time restrictions in LCO 3.0.3 for MODE changes must always be met. This change provides the plant management and operating staff with the flexibility to determine the best course of action should LCO 3.0.3 be entered. This change is consistent with Traveller WSTS-3, C.2. This is an ITS Category (i) change.

3.0Q4 The proposed change to LCO 3.0.3 on page 3.0-1 is a generic ISTS change and requires a staff approved NEI traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-07. [See also comment #231]

3.0Q5 The proposed change to LCO 3.0.3 Bases on page B 3.0-3 is a generic ISTS change that requires a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-07. [See also comment #231]

xvii. The bases for SR 3.0.1 were revised to clarify that credit may be taken for unplanned events that satisfy the performance of an SR. This change allows the deletion of multiple Notes within the SRs in Chapter 3 which state the same thing. The change is consistent with Traveller WSTS-3, C.3: This is an ITS Category (iii) change.

3.0Q6 The proposed change to SR 3.0.1 on page 3.0-9 is a generic change to the ISTS that requires a staff approved traveller.

Response: This change has been submitted as an industry proposed traveller TSTF-08. [This traveller was rejected by the NRC on 12/1/95. Comment #232 was opened to remove all proposed changes.]

xviii. LCO 3.0.4 and the bases were revised to provide greater clarity and consistency with actual Ginna Station practices. First, the details of why exceptions are allowed to LCO 3.0.4 was deleted from the LCO and relocated to the bases. This change provides consistency with LCO 3.0.3 and SR 3.0.2. Second, the bases were revised to provide easier readability. In addition, current Ginna Station operating practices prevent any MODE change, up or down, with inoperable equipment required for the MODE desired to be entered. Therefore, the discussion that LCO 3.0.4 does not prevent MODE changes during a "normal shutdown" conflicts with these practices and was deleted. These are ITS Category (iii) changes.

3.0Q7 The proposed change to LCO 3.0.4 on page 3.0-2 and Bases page B 3.0-5, B 3.0-6, B 3.0-10 are generic changes to the ISTS that requires a staff approved traveller.

Response: This change was made at the request of the Ginna Station PORC to provide additional clarity and consistency with the rest of the LCO

section. The deletion (or relocation) of the last sentence to LCO 3.0.4 (page 3.0-2) was made to provide consistency with LCO 3.0.3. That is, only the statement that "exceptions to the Specification are provided in the individual Specifications" is left in the LCO while additional discussion and clarification is provided in the bases. As written in the NUREG, the phrase "MODES or other specified conditions in the Applicability" is used twice in the sentence proposed to be deleted which makes it very difficult to follow. Comment #163 has been opened to generate a traveller for this change.

With respect to Bases page B 3.0-5, B 3.0-6, and B 3.0-10, the following is provided:

- a. Deletion of last sentence of B 3.0-5 and first sentence of B 3.0-6 - Ginna Station implements LCO 3.0.4 at all times. Consequently, LCO 3.0.4 applies when coming down for a normal shutdown (e.g., the plant is not allowed to enter LTOP conditions with one or both PORVs in operable). This is a conservative change from the NUREG that should be addressed in the final resolution of Traveller BWR-26, C.1. Comment #164 has been opened to track this resolution of this issue from the WOG standpoint.
- b. Addition of text to first paragraph of page B 3.0-6 - This is the relocation of the text from the LCO section as discussed above.
- c. Deletion of text in the second paragraph of page B 3.0-6 - This is an error. Comment #165 has been opened to add this deleted text back into the ITS.
- d. Addition of text to B 3.0-10 - See response to 3.0Q2.

Sections 3.1, 3.2, 3.4 and 3.5 TS

6. Technical Specification 3.1.1

- i. TS 3.1.1.1.b - This requirement was changed to require entry into MODE 1 \leq 8.5% RTP within four hours versus an immediate power reduction under administrative control. This change defines a specific number of hours to reach this condition which provides greater clarity to the operators. The remaining actions as specified by TS 3.1.1.1.b were relocated to LCO 3.4.5 and are discussed in 6.ii below. This is a Ginna TS Category (v.a) change.

3.4Q1 The Completion time of 4 hours discussed above does not agree with NUREG-1431 Rev 1 or Ginna ITS which specifies six hours. Explain this difference.

Response: The "four hours" is a typographical error that should be "six hours" which is consistent with ITS LCO 3.4.4 and NUREG-1431. This error was introduced during the initial draft ITS LCO 3.4.4 that was prepared for plant staff review which used "four hours." However, based on plant staff comments related to the ability to get below

8.5% RTP within four hours, this was changed to six hours in the ITS without the corresponding change to this text. Comment #49 has been opened to correct this error.

- ix. TS 3.1.1.5.a - The lower limit for pressurizer water level (12%) was not added. This lower limit was related to the previous Safety Injection actuation logic which required a coincident low pressurizer level and low pressurizer pressure trip. This logic was modified as a result of IE Bulletin 79-06A (Ref. 45) to eliminate the coincident low pressurizer level trip (Ref. 46) such that the setpoint is no longer used in an UFSAR Chapter 15 accident analysis. Therefore, the low pressurizer water level setpoint is not required. This is a Ginna TS Category (v.b.3) change.

3.4Q2 This change is identified as a less restrictive change, and it does delete a requirement. (Should this change have been made when the physical change to the trip logic was made?) Provide an explanation as to why this change is a less restrictive.

Response: By definition, any CTS requirement which has been deleted or otherwise relocated outside TS is a "less restrictive" change since there is no longer any TS requirement. With respect to the physical change to the trip logic, this CTS requirement should have also been eliminated at the time of the modification. However, plant procedures still require operators to verify that the pressurizer level is > 12% above 350°F such that the actual requirement is still being met even though it is not necessary to support any safety function.

- xii. TS 3.1.1.3.a and 3.1.1.3.b - These requirements were not added to the new specifications since the pressurizer safety valves do not provide overpressurization protection during Cold Shutdown and Refueling conditions. This is provided by the low temperature overpressure protection (LTOP) requirement as specified in current TS 3.15 and new LCO 3.4.12. Since the pressurizer safety valves do not perform a safety function during these low MODES of operation, these requirements were not retained. These changes also supersede those proposed in Reference 60. This is a Ginna TS Category (v.b.4) change.

3.4Q3 Provide an explanation for this being a less restrictive change. You state that the pressurizer safety valves never performed the safety function for protecting the RCS from low temperature overpressurization, yet the valves were there (operable) in the lower MODES, why then could these valves not protect the RCS from high pressure at low temperature? What is the importance of using Reference 60 in the discussion of change justification?

Response: Since the requirement for the pressurizer safety valves in Cold Shutdown and Refueling is being removed from the CTS, this is a "less restrictive" change with respect to current requirements. The LTOP System is designed to actuate at 424 psig to maintain the RCS less than 110% of the RHR System design pressure of 600 psig (see ITS LCO 3.4.12). Consequently, the RCS will never reach the pressurizer safety valve lift setpoint of 2485 psig ± 1% as required

by CTS 3.1.1.3.a. It should be noted that the LTOP System was added after initial startup for Ginna Station such that the pressurizer safeties were originally relied upon for pressure reduction during lower MODES. With respect to Reference 60, RG&E had originally proposed changes to the CTS requirements for the pressurizer safety valves and PORVs in response to Generic Letter 90-06. The changes proposed in the May 26, 1995 submittal were identified as superseding those TS changes.

- xvii. TS 3.1.1.3.c - This was revised to change the pressurizer safety valve lift settings from 2485 psig \pm 1% to 2485 psig + 2.4%, -3%. The valve lift settings are required to be set to within \pm 1% following testing; however the OPERABILITY tolerances have been revised. The increased OPERABILITY tolerances have been evaluated in the most limiting pressure transients for Ginna Station (i.e., loss of external load and locked rotor events) and found to result in acceptable results with respect to the safety limit values. This change is a result of an event in which the pressurizer safety valves were found to have drifted outside the existing \pm 1% tolerance band following testing (Ref. 58). Revising the OPERABILITY tolerances will reduce the potential for future LERs for an issue which has been demonstrated to remain within the accident analysis requirements. This is a Ginna TS Category (v.b.45) change.

3.4Q4 This justification should be separated from reducing the frequency of LERs. This is not a safety analysis basis for changing the tolerances. The argument on the transient analysis results has merit and should be expanded to indicate how these values meet code setpoint requirements. Provide the details of the analysis and any explanation of why this change is considered within the scope of conversion to the ISTS.

Response: The safety analysis basis for revising the pressurizer safety valve's tolerance band is summarized in the letter from Westinghouse to RG&E dated September 8, 1995 (attached). The ASME code tolerance requirements for relief valves set above 1000 psig are \pm 1% following testing and \pm 3% for OPERABILITY (see NUREG-1431, bases for SR 3.4.10.1). The proposed changes are within these ASME limits. This change is being performed in conjunction with the ISTS submittal and was identified by RG&E as being an issue when we committed to perform this conversion to ISTS due to a recent LER on this issue. RG&E also agrees to delete the second to last sentence of the change justification related to LERs. Comment #49 has been opened to address this.

7. ii. TS 3.1.2.1.b - The requirement for periodically recalculating the RCS temperature and pressure curves and the RCS heatup and cooldown curves and limits was relocated from technical specifications to the PTLR. A periodic review is already required by 10 CFR 50, Appendix H which does not need to be restated within the technical specifications. This is a Ginna TS Category (iii) change.

3.4Q5 The PTLR itself does not specifically address the requirement to periodically recalculate the heatup and cooldown curves and limits. Identify the location of this requirement in the improved TS even

though this requirement may be contained in some of the referenced material. (As stated above, a periodic surveillance program is required by 10 CFR 50, Appendix H which specifies requirements for specimen withdrawals and for determining if TS need to be changed, but Appendix H is non-specific on that point.)

Response: CTS 3.1.2.1.b requires that the "limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using the methods discussed in the Basis Section." These two figures are developed based on reactor vessel capsule surveillances and other factors (e.g., weld chemistry) such that figures are valid for a specified effective full power years (EFPYs). Unless any of the factors which are used to generate the figures have changed, or the figures are nearing the end of their specified applicability, there is no need to recalculate them. 10 CFR 50, Appendix H requires a schedule for withdrawing surveillance capsules, and subsequent to their withdrawal, requires a determination if the two subject figures need to be revised. All of the remaining factors used to calculate these two figures are not expected to change. Consequently, RG&E has interpreted CTS 3.1.2.1.b to require recalculation based on the surveillance requirements of Appendix H. RG&E agrees that the change justification is incorrect in stating that this periodic recalculation is being relocated to the PTLR since it is, in fact, being deleted as it is duplicating existing regulations. Comment #49 has been opened to remove the statement "relocated from technical specifications to the PTLR" and replace it with "deleted from technical specifications."

iv. TS 3.1.2.2 - This was not added since this temperature limit is not required for safe operation. All necessary heatup and cooldown rates are relocated to the PTLR while new LCO 3.4.1 provides limits on RCS pressure, temperature, and flow. This is a Ginna TS Category (v.b.5) change.

3.4Q6 This change is not adequately justified; i.e., "not required for safe operation" is not adequate. Provide justification for deletion of this requirement.

Response: The bases for CTS 3.1.2.2 state that "the temperature requirements for the steam generator corresponds with the measured NDT for the shell of the steam generator." RG&E is replacing steam generators in the spring of 1996 with new SGs that do not have this concern. In addition, the CTS only applies during MODE 5 or 6 since this is the only time in which a steam generator can be < 70°F at Ginna Station as there are no RCS isolation valves. That is, the steam generator vessel cannot be maintained < 70°F and < 200 psig with the RCS greater than 200°F unless the steam generator was physically isolated from the RCS heat source. Therefore, with the RCS at reduced temperature and pressure, the consequences of faulting the steam generator under these conditions is significantly reduced.

8. i. TS 3.1.3.1 - This was revised to raise the minimum temperature for criticality from 500°F to 540°F. This change was made to correct a discrepancy between the definition of reactor operating modes and this requirement. Currently, Ginna Station TS 1.2 defines Hot

Shutdown as Reactivity $\leq -1 \Delta k/k\%$ and $T_{avg} \geq 540^\circ F$. In order to achieve criticality at $500^\circ F$, the Hot Shutdown condition would have to be directly bypassed. A value of $540^\circ F$ was selected for the new minimum temperature for criticality based on previous operating experience during startup conditions. This is a Ginna TS Category (v.a) change.

- 3.4Q7 Provide additional justification to explain how "based on previous operating experience" relates to the new minimum temperature for criticality and relate it to operating temperature, if appropriate.

Response: Revising the minimum temperature for criticality from $500^\circ F$ to $540^\circ F$ in CTS 3.1.3.1 is a conservative change since it requires additional RCS heatup before obtaining criticality. CTS 1.2 does not specify at which point the reactor goes critical, only that Operating temperature is " $\sim 580^\circ F$ " and that Hot Shutdown is " $\geq 540^\circ F$." Normal operating temperature for Ginna Station is actually $573.5^\circ F$. The temperature at which criticality is typically reached is approximately $545^\circ F$; however, allowing for instrument uncertainty, this could actually be as low as $540^\circ F$. Hot Zero Power (HZP) at Ginna Station is $547^\circ F$ at which temperature several safety analyses are performed. The proposed $7^\circ F$ tolerance does not adversely affect any of these safety analyses since the MTC is not significantly affected by this small temperature difference as discussed in the bases for ITS LCO 3.4.2.

- iii. TS 3.1.3.3 - The existing action statement was revised to require that the plant be in MODE 2 with $k_{eff} < 1.0$ within 30 minutes if T_{avg} for one or both RCS loops was $< 540^\circ F$ versus subcritical by an amount equal to or greater than the potential reactivity due to depressurization. The new requirement provides clear and precise instructions to operations and ensures that the plant is quickly brought to a condition in which the LCO is no longer applicable. This is a Ginna TS Category (v.c) change.

- 3.4Q8 Provide an explanation for why it is acceptable to delete the requirement to insert sufficient negative reactivity to offset the effects of depressurization since this is significantly more reactivity than required to simply insert enough reactivity to become just subcritical.

Response: ITS LCO 3.4.2 Required Action A.1 specifies that if T_{avg} falls below $540^\circ F$, the reactor must be brought to a subcritical condition within 30 minutes. Once subcritical conditions are reached (i.e., MODE 2 with $k_{eff} < 1.0$), ITS 3.1.1 requires that Shutdown Margin (SDM) limits shall be met. This LCO ensures that sufficient reactivity is available to offset the effects of depressurization as required in CTS. Therefore, the CTS are ITS are essentially equivalent.

- v. TS 3.1.3.1 - This was revised to reference cycle specific MTC requirements in the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The MTC maximum upper limit described in TS 3.1.3.1 remains the same in ITS LCO 3.1.4. This is a Ginna TS Category (iii) change.



3.4Q1 Explain what elements are relocated to the COLR program and what elements are relocated outside the TS to owner controlled documents.

Response: CTS 3.1.3.1 contains requirements for: (1) minimum temperature for criticality; (2) MTC upper limit when at HZP and below 70% RTP; and (3) MTC upper limit when at HZP and above 70% RTP. The minimum temperature for criticality is contained in ITS LCO 3.4.2 as discussed above in response to 3.4Q7. The two MTC upper limits at HZP are both specified in ITS LCO 3.1.4. The MTC beginning of cycle life (BOL) and end of cycle life (EOL) lower limits, which are not in CTS, are specified in the COLR (see Attachment F to the 5/26/95 submittal). There is no MTC limit relocated from CTS to an owner controlled document.

9. i. TS 3.1.4.4 - This specification was revised to only require shutdown to MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 8 hours versus Cold Shutdown within 40 hours consistent with the LCO Applicability. This is a Ginna TS Category (v.c) change.

3.4Q9 Provide an explanation discussing how this is a (v.c) change since the revised LCO does not require going to cold shutdown and existing TS do.

Response: Once the plant has reached $T_{avg} < 500^{\circ}\text{F}$, the I-131 equivalent activity limit of CTS 3.1.4.1 no longer applies. Therefore, the plant has exited the Mode of Applicability for this requirement and further shutdown to Cold Shutdown as specified in CTS 3.1.4.4 is not required. Since there is no technical change in the response to exceeding I-131 equivalent activity limits in the conversion from CTS to ITS, this was identified as an "administrative change."

10. i. TS 3.1.5.1.1 - Added a new requirement for the containment sump "A" level or pump actuation per LCO 3.4.15. This leakage detection system replaces the containment humidity detectors and the air cooler condensate flow monitor. The containment humidity detectors do not meet the required leakage rate detection capability of 1.0 gpm within 4 hours as required by Generic Letter 84-04 (Ref. 19). In addition, the containment humidity detectors are recommended by RG 1.45 (Ref. 17) to only be used as an alarm or indirect indication of leakage to containment and not as a separate method of detecting leakage. The remaining leakage detection systems provide adequate monitoring as discussed in the new bases and Section C, item 46. These are Ginna TS Category (v.a) changes.

3.4Q10 Provide a brief explanation of the sump pump monitor alarms. Discuss whether the alarms actuate on pump actuation frequency, total pumping time, and sump level, and discuss how leakage is quantified.

Response: The sump pump monitor alarms actuate on closing of the sump pump breaker and control room annunciators remain lit until the pump stops and the breaker opens. The operation of the pump is logged by control room operators such that a sump pump actuation interval and associated leakage rate is determined every shift. Copies of the

two procedures which require logging of the sump pump actuation (AR-C-18) and the determination of sump pump interval and leakage rate (S-12.4) are attached.

- iv. TS 3.1.5.2.2.c - The requirement to commence a reactor shutdown with excessive SG tube leakage was revised to allow an additional 4 hours to correct administrative and other similar discrepancies in the Steam Generator Tube Surveillance Program consistent with LCO 3.4.13.B. Requiring a reactor shutdown for most administrative errors is not prudent based on the increased risk for a transient while changing MODES. However, if the integrity of the tube is determined to be inadequate, a reactor shutdown will continue to be immediately initiated. Also, the requirement to perform a SG inspection with excessive leakage if an inspection has not been performed within the last 6 months was not added to the new specifications. Any SG inspections will be determined as part of the corrective actions necessary to repair the leaking tube and in accordance with the Steam Generator Tube Surveillance Program. Since LCO 3.0.4 applies to this LCO, the plant cannot go above MODE 5 without verifying that the SG tube integrity is acceptable. These are Ginna TS Category (v.b.8) changes.

3.4Q11 The reason for adding Condition B according to the Bases is, "to allow an additional 4 hours to correct administrative and other similar discrepancies in the Steam Generator Tube Surveillance Program . . .". The reason for Condition B is not clear. It should not be associated with SG leakage as is done in this justification. The lead-in sentence to this item states that "the requirement to commence a reactor shutdown with excessive leakage was revised to allow an additional 4 hours" is not associated with steam generator leakage. Explain how the proposed action to declare the SG inoperable if there is a program deviation is relevant to LCO 3.4.13 requirements? Provide a better explanation and safety basis justification on the need for Condition B.

Response: Condition B was added as a result of approved Traveller WOG-15, C.1, Revision 1 (attached). The justification for this traveller was based on the desire to address steam generator program deficiencies found while in MODES 1, 2, 3, or 4 to allow licensees time to either correct the problems or contact the NRC. The intention of the RG&E justification was to state that CTS 3.1.5.2.2.c was being revised to also address steam generator program deficiencies. For cases where deficiencies in the program exist, but where primary to secondary LEAKAGE is still within allowed limits, 4 hours is provided to correct the administrative problem. Comment #49 has been opened to clarify this RG&E change justification. [This response was revised per comment #239]

3.4Q12 Justify changing the existing TS requirement from "be at hot shutdown within 6 hours and at an RCS temperature less than 350°F within the following 6 hours" to "Be in MODE 3 in 6 hours and "Be MODE 5 in 36 hours."

Response: The RCS LEAKAGE limit MODE of Applicability have been expanded from above 350°F (or MODE 3) to above 200°F (or MODE 4) which is a

conservative change consistent with NUREG-1431. Consequently, if RCS LEAKAGE limits are not met, the plant must enter MODE 5 to exit the MODE of Applicability for ITS LCO 3.4.13. Currently, the plant must only go to MODE 4 to exit the MODE of Applicability and has 12 hours to perform this action. Allowing an additional 24 hours to reach MODE 5 is considered acceptable due to the time required to reach these conditions. The time to reach MODE 5 is also consistent with LCO 3.0.3.

3.4Q13 Justify why and how the Steam Generator Tube Rupture Program replaces the requirement to perform a SG inspection if one has not been performed in last 6 months.

Response: CTS 3.1.5.2.2.c requires that with excessive steam generator leakage, the plant must shutdown and "If more than six months have elapsed since the last steam generator inspection, perform an inspection in accordance with the requirements of Technical Specification 4.2." CTS 4.2.1.4 only states that inspection intervals for steam generator tubes shall be specified in the IST Program and provides eddy current detected tube imperfection acceptance limits. Consequently, the IST Program really contains the inspection intervals for both normal testing requirements and following detection of excessive leakage. Removing this statement from CTS 3.1.5.2.2.c is acceptable since CTS 4.2 does not have any actual testing requirements.

12. i. TS 3.2.5 - The requirement was revised to require placing a charging pump in pull-stop within 1 hour regardless of the status of the RHR pumps or the MODE. This is a conservative change which provides direct operator guidance to perform an action within a defined time period. Also, these requirements were relocated to the LTOP specification to consolidate all related requirements. The verification of the charging pump status every 12 hours was also not added since the plant is required to be in a depressurized and vented condition within 8 hours which removes the need to isolate a charging pump (i.e., a 1.1 square inch vent can mitigate a charging/letdown mismatch event). These are Ginna TS Category (v.a), (i), and (v.c) changes, respectively.

3.4Q14 Provide appropriate design basis and safety analysis basis justification why Ginna needs only to protect against a single charging pump starting for vented RCS conditions when there are 3 operable charging pumps.

Response: The requirement to only isolate one charging pump for RCS vented conditions in ITS LCO 3.4.12 is consistent with CTS 3.2.5. The NRC SER which implemented CTS 3.4.12 is attached. Although this SER does not specifically address the charging pump issue, it references an RG&E letter dated February 24, 1977 (as do the bases for CTS 3.2.5). This letter (attached) provides the results of an analysis demonstrating that relief valve 203 in the RHR System can mitigate 2 of 3 charging pumps following complete isolation of letdown. The internal flow path diameter of relief valve 203 is 0.5 square inches which is less than the CTS and ITS required 1.1 square inch vent path.

13. i. TS 3.3.1.1.b and 3.3.1.3 - LCO 3.5.1 Condition A was added which allows 72 hours to restore accumulator boron concentration to within acceptable limits. The ITS bases state that allowing a longer period of time to correct boron concentration is acceptable since the volume of water in the accumulators is the critical feature. Attempting to correct boron concentration within the current 1 hour limit would create a significant burden on the operations staff. Therefore, the current 1 hour LCO was only maintained for accumulator pressure and volume. In addition, the accumulator boron concentration limits were relocated to the COLR since these values can change due to refueling cycle changes. These are Ginna TS Category (v.b.9) and (iii) changes, respectively.

3.5Q1 Provide additional justification for the change from a 1 hour to 72 hours TS limit to adjust accumulator boron concentration by documenting the operational hardship with the typical time needed to complete a boron adjustment and to achieve an acceptable concentration. Also, include any safety analysis basis that supports the proposed change.

Response: Once it is determined that the accumulator boron concentration needs to be changed, operations and chemistry must first calculate the required boron to be added. The accumulator is then drained to just above the TS minimum required level and the borated water added from the RWST. Depending on the initial accumulator boron concentration, the tank may need to be drained and filled several times to reach the required limit since the RWST is maintained at only 2000 ppm and there is only 25 cubic feet of water volume in the accumulator to work with. Since these actions can take longer than one hour to complete, RG&E attempts to maintain accumulator boron concentration sufficiently above the TS limit such that actions are initiated before TS limits are reached. This is performed in part by checks of accumulator level every shift. With respect to the accident analysis for Ginna Station, the water inventory in the accumulators is the most significant feature due to the boron concentration available from the RWST. The critical accident scenario with respect to boron concentration is a steam line break for which the accumulators do not dump.

- vii. TS 3.3.1.1.b - The bases for TS 3.3 were revised to update the specified water volume contained in the accumulator with respect to the 50% and 82% levels. The required levels specified in TS 3.3.1.1.b have not been changed, only the corresponding water volumes provided in the bases. The new values are consistent with those used in the accident analysis (see COLR, Table 1). This is a Ginna TS Category (v.c) change.

3.5Q2 The Bases for ITS LCO 3.5.1, Accumulators, do not provide the accumulator volume assumed in the accident analysis as stated. Also, could not find this value in existing TS 3.3 Bases as indicated. Provide an explanation that describes how operators will have access to these volumes.

Response: CTS bases page 3.3-13 contains the accumulator water volumes. This states that the accumulator 50% indicated level is equivalent to

1108 cubic feet of water while the 82% indicated level is equivalent to 1134 cubic feet of water. ITS SR 3.5.1.2 states that 50% equates to 1126 cubic feet while 82% equates to 1154 cubic feet. These changes are based on more recent calculations of the actual accumulator design and are conservative changes with respect to the accident analyses (i.e., the analyses were performed assuming only 1108 cubic feet and 1134 cubic feet were available). Since these volumes are actually specified in the SR, they are not reiterated in the bases for ITS LCO 3.5.1. With respect to operators access to these volumes, as long as the accumulators are maintained within the required indicated levels of ITS SR 3.5.1.2, this will ensure that the necessary volume as assumed in the accident analysis is available.

20. xii. TS 3.10.4.2 and TS 3.10.4.3 - These were revised to remove conditions of rod inoperability due to being immovable. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if they are immovable. Reference to full length rods was also removed since there are no part length rods in the reactor core. This is a Ginna TS Category (v.c) change.

3.1Q2 Provide a safety basis explanation describing how the change to redefine rod OPERABILITY is an administrative change, especially since the existing TS would require an immovable rod to be declared inoperable and put the plant on a shutdown track.

Response: CTS 3.10.4.2 requires a control rod to be declared inoperable due to "being immovable as a result of excessive friction or mechanical interference or known to be untrippable." CTS 3.10.4.3 provides requirements for a control rod "inoperable for causes other than addressed by 3.10.4.2, above, or misaligned from its group step counter demand position by more than ± 12 steps." RG&E has always interpreted CTS 3.10.4.2 to address problems in which the control rod was unable to insert into the core within accident assumed time limits (i.e., that the rod was "untrippable"). Meanwhile, CTS 3.10.4.3 addresses problems such as a control rod which is unable to fully withdraw. In this case; CTS require aligning all rods in the affected control rod group or reducing power to $\leq 75\%$ RTP within 1 hour.

RG&E has proposed to remove "being immovable as a result of excessive friction or mechanical interference" from CTS 3.10.4.2 and "inoperable for causes other than addressed by 3.10.4.2, above" from CTS 3.10.4.3. The bases for ITS LCO 3.1.4 state that "if a control rod(s) is discovered to be immovable but remains trippable and aligned, the control rod is considered OPERABLE." This bases discussion is consistent with RG&E's interpretation of these CTS requirements such that the deleted CTS text is only considered an administrative change since the same requirements are being implemented.

- xiii. TS 3.10.4.3.2 - This was revised to remove the requirement to declare a misaligned rod inoperable when the rod cannot be restored to within the alignment limits in 1 hour. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if

they are immovable. This is a Ginna TS Category (v.a) change.

- 3.1Q3 Provide an safety basis explanation describing how the change to define a misaligned rod to be OPERABLE, even if unable to restore alignment within one hour, is a more restrictive change. Existing TS would require a misaligned rod to put the plant on a power reduction track unless restored to within limits in one hour.

Response: CTS 3.10.4.3 addresses misaligned control rods and requires that it be restored to OPERABLE status or the rod declared inoperable and shutdown margin (SDM) verified within 1 hour. In addition, if the control rod is declared inoperable, the remaining rods in the affected control group must be either aligned with the inoperable control rod or power reduced to $\leq 75\%$ RTP within 1 hour. RG&E has proposed to delete the requirement to declare the rod inoperable if it is not restored to OPERABLE status within 1 hour since if the option to align all rods in the affected control group is chosen, the rod is no longer inoperable because it is within alignment limits in both CTS and ITS. If the option to reduce power to $\leq 75\%$ RTP is chosen, several additional actions are required including a verification that the accident analyses remain valid with the misaligned rod. Once this accident analysis verification has been completed, operation may continue but the rod remains inoperable. Therefore, in the first of these two options, the affected control rod is not inoperable and deleting this text is actually an "administrative change" instead of a "more restrictive change." Comment #50 has been opened to correct this error in Attachment A of the submittal.

- xvi. TS 3.10.4.3.2.b and TS 3.10.4.3.2.c - These were revised to remove the requirement to reduce the high neutron flux trip setpoint to $\leq 85\%$ RTP when the power level is reduced to $\leq 75\%$ RTP. This required action is deleted based on agreements between the NRC and the owners groups and is consistent with WCAP-13029 (Ref. 50) which states that the safety analyses results would not be significantly affected by changes to their initial assumptions as a result of increased peaking factors caused by rod misalignment. Additionally, the peaking factor limit verification within 72 hours and the re-evaluation of the safety analysis within 5 days that are required by this specification provide further assurance that the assumptions made in the safety analysis are preserved. This is a Ginna TS Category (v.c) change.

- 3.1Q4 Provide an safety basis explanation as to why this change is an administrative change since the trip setpoints are no longer required to be reset downward. Resetting downward will provide an earlier trip which is in the conservative direction.

Response: RG&E agrees that this is not an administrative change and is actually a "less restrictive change." Comment #51 has been opened to address this issue, including the need for a no significant hazards evaluation.

- xvii. TS 3.10.4.4 - This was revised to include an action to verify SHUTDOWN MARGIN or initiate boration within 1 hour when more than

one rod is out of alignment. The ITS Bases state that 1 hour is a reasonable time based on the time required for potential xenon distribution and the low probability of an accident. This is a Ginna TS Category (v.a) change.

3.1Q5 Provide a rationale for the more restrictive change, (v.a) category applied to this item. It appears to be less restrictive since the new requirement allows an hour to determine SDM or to borate and restore the margin if need be, but still requires the plant to be in hot shutdown in 6 hours.

Response: This is a less restrictive change since even though CTS and ITS require the plant to be in MODE 2 with $k_{eff} < 1.0$ (or Hot Shutdown) within 6 hours in this condition, the ITS also require verification of SDM within the first hour. Consequently, with the additional SDM verification required in ITS, this is a "more restrictive" change with respect to CTS requirements.

xviii. TS 3.10.5.1 - This was revised to add an action statement to clarify that if more than one MRPI is inoperable per group for one or more groups or more than one demand position indicator per bank is inoperable for one or more banks then the plant must enter 3.0.3 immediately. This is a Ginna TS Category (v.a) change.

3.1Q6 NUREG-1431 does not have a specific Condition for entry into LCO 3.0.3 as ITS LCO 3.1.7, Rod Position Indication does, but it is implicit from the generic LCO 3.0.3 discussion of requirements that LCO 3.0.3 would apply if the condition exists and the specific LCO does not so state it. Explain the reason for inclusion of statement to enter into LCO 3.0.3.

Response: At the request of the Ginna Station Operations department, almost every LCO has a Condition which addresses entry into LCO 3.0.3. This was done to assist in the operational transition from custom TS to ITS and as a human factors consideration. That is, several LCOs must have a Condition which addresses entry into LCO 3.0.3 due to design considerations (e.g., for a system with four different trains, the loss of more than two trains may be a loss of safety function as shown in NUREG-1431, LCO 3.5.1)). Specifying this LCO 3.0.3 entry in almost every LCO instead of only a limited number provides operators with very clear and concise directions. In addition, with respect to ITS LCO 3.1.7, the rod position indication system used at Ginna Station is different from that provided in NUREG-1431. The CTS are also very clear in that no more than one MRPI per group may be inoperable and no more than one demand position indicator per bank may be inoperable at any one time.

xx. TS 3.10.2.1 - This was revised to require measurement of the power distribution after each fuel reloading prior to operation of the plant at or above 75% RTP instead of prior to 50% RTP consistent with ITS. This requirement ensures that the design limits are not exceeded when RTP is achieved, since peaking factors are usually decreased as power increases. Requiring this surveillance at 75% versus 50% still provides the necessary margin to ensure that design

safety limits are not exceeded and provides the operator with more flexibility during power ascension following a refueling. This is a Ginna TS Category (v.b.25) change.

- 3.2Q1 Provide documentation of the conclusions for why this less restrictive change is acceptable. What is the safety basis explanation for the proposal to ascend to 75% RTP without confirming that fuel design safety limits are not exceeded? Explain the statement, "since peaking factors are usually decreased as power is increases" and what makes this a factual statement.

Response: The original requirement to measure the power distribution prior to 50% RTP during plant startup was based on engineering judgement and the desire to verify, prior to reaching RTP, that the design limits will not be achieved. Current industry practice associated with this technical specification requirement has since been revised to allow plants to measure the power distribution prior to 75% RTP during plant startup in order to provide operator flexibility associated with the xenon oscillations prevalent to the lower power levels which may affect the power distribution measurements. Performing the verification of power distribution limits prior to 75% RTP versus 50% RTP is consistent with NUREG-1431 SR 3.2.1.1 and SR 3.2.2.1. RG&E requested that Westinghouse confirm that the increase to 75% RTP was acceptable for Ginna Station. Westinghouse confirmed that this increase was acceptable and provided the above change justification.

There are several requirements outside of technical specifications which involve the performance of flux maps and power distribution measurements at lower power levels during startup to satisfy Physics Testing requirements. These measurements are currently performed at 30% and 60% at Ginna Station. As such, while multiple flux maps will be performed at various power levels during startup, verifying peaking factors are within technical specification requirements at 75% RTP will allow the plant to achieve a xenon equilibrium condition which will provide a more accurate indication of the actual power distribution. The peaking factors are a function of the power level and at reduced power levels, the peaking factor limits are increased based on the associated power multiplication factor. Therefore, the closer the plant is to RTP conditions during the performance of the power distribution measurements, then the more meaningful the measurement will be with respect to actual RTP conditions.

- xxv. TS 3.10:2.2 - This was revised to allow 72 hours (instead of 24 hours) to reduce the Overpower ΔT and the Overtemperature ΔT trip setpoints when F_Q or $F_{\Delta H}$ is not within limits consistent with NUREG-1431. This section was also revised to include a Completion Time of 72 to reduce the Power Range Neutron Flux High trip setpoints. These actions provide further protection against the consequences of severe transients with unanalyzed power distributions. The 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the initial prompt reduction in THERMAL POWER. This is a Ginna TS Category (v.b.27) change.

3.2Q2 Provide an safety basis explanation to further justify the change from 24 hours to 72 hours to adjust these setpoints when the hot channel factor limits are exceeded.

Response: The 72 hours to reduce the Overpower ΔT and Overttemperature ΔT trip setpoints is consistent with NUREG-1431. Providing an additional 48 hours to reduce these trip setpoints is acceptable since the prompt reduction in THERMAL POWER within 15 minutes provides the necessary compensatory actions when hot channel factor limits are exceeded. Revising the trip setpoints is a conservative action beyond this initial action. The additional 48 hours also provides greater time to plan and prepare the trip setpoint reduction to reduce the likelihood of a reactor trip during this activity. Revising the Power Range Neutron Flux High trip function is also consistent with the above discussion but is being addressed under an industry traveller. Comment #53 has been opened to track this traveller.

28.i.b 6. SR 3.1.6.1 - Requires verification within 4 hours prior to criticality that the critical control bank position is within limits in the COLR.

3.1Q7 Provide an explanation why the "within 4 hours" is not included in the Surveillance Requirement statement or the Frequency.

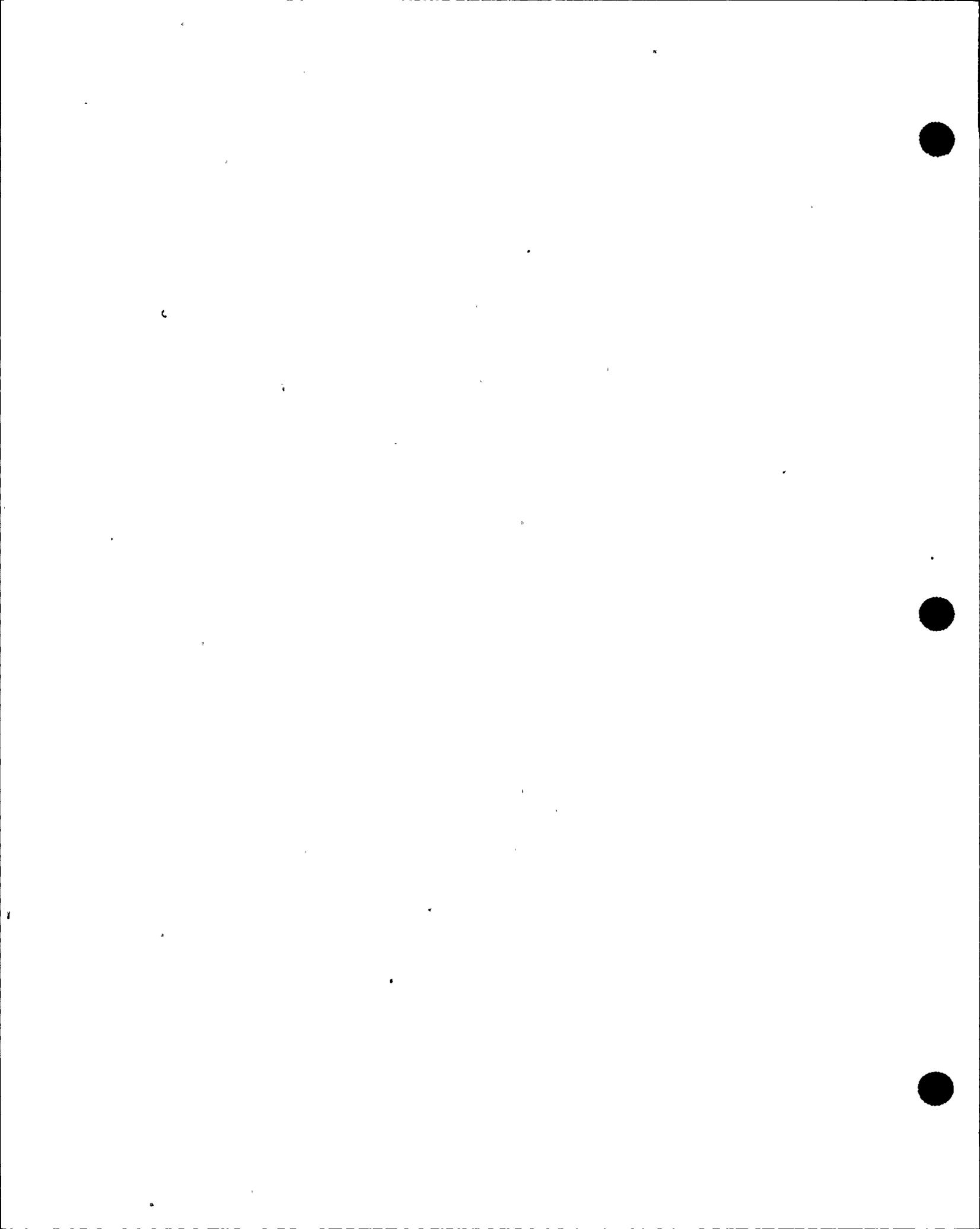
Response: The "within 4 hours" is a typographical error and should be removed from the change justification. The reason for removing the "within 4 hours" from NUREG-1431 SR 3.1.7.1 is provided in Attachment A, Section C, item 14.xii. Comment #50 has been opened to correct the Attachment A typographical error. [This response was revised based on meeting week of 9/18/95. See comment #70.]

8. SR 3.1.8.4 - Requires verification every 30 minutes during MODE 2 PHYSICS TESTS that THERMAL POWER \leq 5% RTP. Verification of the THERMAL POWER level will ensure that the initial conditions of the safety analyses are not violated.

3.1Q8 The SR number in the ITS is SR 3.1.8.3 instead of SR 3.1.8.4 as stated above. Provide a confirmation of the correct number.

Response: This is a typographical error in Attachment A which should state SR 3.1.8.3. Comment #50 has been opened to correct this error.

28.iv.a. Table 4.1-4, Functional Unit #1 was revised as indicated in ITS SR 3.4.16.1 to only require verification of reactor coolant gross specific activity once every 7 days when $T_{avg} \geq 500^{\circ}F$ versus once every 72 hours above Cold Shutdown (i.e., $T_{avg} \geq 200^{\circ}F$). The increased surveillance interval is acceptable based on the small probability of a gross fuel failure during the additional 4 days. Fuel failures are more likely to occur during startup or fast power changes and not during steady state power operation during which the majority of sampling is performed. Gross fuel failures will also result in Letdown radiation alarms and possibly containment radiation alarms providing additional operator indication. Only requiring this surveillance when $T_{avg} \geq 500^{\circ}F$ provides consistency



with the LCO Applicability. This is a Ginna TS Category (v.b.34) change.

3.4Q15 The decrease in the activity determination interval needs supporting justification. Provide documentation concerning how the alarms will protect the activity limit during this increased surveillance interval.

Response: The Letdown System contains a radiation alarm (R-9) which is an ion chamber detector with a range of 0.1 mr/hr to 10,000 r/hr specifically installed to detect possible fuel failures. Both indication and alarm for R-9 are available in the control room. This monitor, has in fact, detected fuel problems in the past and is checked every shift by control room operators.

28.iv.c Table 4.1-4, Functional Upit #3 was revised per ITS SR 3.4.16.3 to delay determination of E until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation following the reactor being subcritical for ≥ 48 hours. The 31 days was added to ensure that radioactive materials are at equilibrium in order to provide a true representative sample for E determination and eliminate possible false samples. This is a Ginna TS Category (v.c) change.

3.4Q16 Explain why this is an administrative change, since it changes the sampling and analysis technical requirements.

Response: RG&E agrees that this is not an administrative change and is actually a "less restrictive change." Comment #52 has been opened to address this issue, including the need for a no significant hazards evaluation.

30. iv. TS 4.3.3.1, 4.3.3.2, and 4.3.3.3 - The requirement that the leakage tests be performed with a minimum test differential pressure of 150 psid was not added to the new specifications. The Bases for new LCO 3.4.14 reference ASME, Section XI (Ref. 53) which provides acceptable guidance for performing these leakage tests. This includes adjusting the observed leakage rates for tests that are not conducted at the maximum differential pressure by assuming that leakage is directly proportional to the pressure differential to the one-half power. This is a conservative change in most cases since it requires that the PIVs be tested under the maximum differential pressure conditions. This is a Ginna TS Category (v.c) change.

3.4Q17 Provide a safety basis explanation supporting the statement that the proposed change is conservative in most cases since PIVs will be tested under maximum differential pressure conditions by adjusting observed leakage rates for tests not conducted at the maximum differential pressure. Identify those cases where the results would not be conservative when they are conducted at less than the maximum pressure differential.

Response: The CTS require that leakage testing be performed at greater than 150 psid for all PIVs. The ASME testing requirements as contained in NRC approved standard OMa-1988 (attached), require PIV leakage

testing to be performed with the "full maximum function pressure differential." Leakage tests performed at lower pressures "are permitted in those types of valves in which service pressure will tend to diminish the overall channel opening, as by pressing the disk into or onto the seat with greater force." Therefore, the PIVs are to be tested using the assumed worst case pressure differential unless this causes the valve to seat tighter in which case a lower pressure differential is allowed with a corresponding leakage limit reduction. As such, there is no reduction in safety since the current 150 psid required minimum may in fact be assisting the valve in closing and meeting its leakage acceptance limits.

- v. TS 4.3.3.4 Specifies requirements for allowable leakage rate limits for PIVs. The allowed leakage rates for PIVs was adjusted from a single value for all valves to a value based on valve size consistent with SR 3.4.14.1 and SR 3.4.14.2. This change provides greater information of valve degradation and removes an unjustified penalty on larger valves (Ref. 54). This is a Ginna TS Category (v.c) change.

3.4Q18 Provide an safety basis explanation for why it is acceptable to change the allowable leakage limit to one based on valve size, and explain how this compares with the leakage limit in existing TS.

Response: *The basis for using leakage limits based on size is contained in NRC approved standard OMA-1988 (attached) which is consistent with the EG&G Report referenced in the change justification above. The CTS specify leakage rates less than 1.0 gpm but allow leakage rates up to 5 gpm provided that the following is met:*

$$\frac{(\text{current test leakage rate}) - (\text{previous test leakage rate})}{0.5} <$$

$$(5.0 \text{ gpm}) - (\text{current test leakage rate})$$

Specifying leakage rates based on size (0.5 gpm per valve diameter) results in the following new acceptance criteria:

- a. *Check Valves 878G and 878J (CTS 4.3.3.2) and 877A, 877B, 878F, and 878H (CTS 4.3.3.3) are 2 inch valves such that their new leakage criteria will be 1 gpm. Therefore, this is a less restrictive change since leakage rates up to 5 gpm are allowed for these valves.*
- b. *Motor operated valves 878A and 878C (CTS 4.3.3.3) are 2 inch valves such that their new leakage criteria will be 1 gpm. Therefore, this is a less restrictive change since leakage rates up to 5 gpm are allowed for these valves.*
- c. *Check valves 853A and 853B (CTS 4.3.3.1) are 5 inch valves such that their new leakage criteria will be 2.5 gpm. Check valves 867A and 867B (CTS 4.3.3.1) are 10 inch valves such that their new leakage criteria will be 5.0 gpm. Both of these leakage rates are within the CTS limit. In addition, these valves only provide the first isolation barrier to the RCS since there is at least one other valve in series with each of these valves which must fail to allow RCS pressurized fluid into the RHR and SI Systems.*

32. iii. TS 4.5.2.2.c - The test related to accumulator check valve testing for operability every refueling shutdown was relocated to the Ginna

Station Inservice Testing program. The valves are currently partially stroke tested quarterly and refurbished every six years. Leakage associated with these check valves is addressed by SR 3.5.1.2. This is a Ginna TS Category (iii) change.

3.5Q3 Provide an explanation of how the leakage past these valves is addressed by ITS SR 3.5.1.2.

Response: ITS SR 3.5.1.2 requires verification of accumulator volume every 12 hours. The SI pumps are tested monthly by use of a test line to the RWST (see attached UFSAR drawing 6.3-1, Sheet 2). Since the SI pumps are only designed with a shutoff head of approximately 1400 psig, the injection lines are not isolated from the RCS other than by the normally closed check valves. The accumulators are maintained between 700 psig and 790 psig. Therefore, if accumulator check valves 842A and 842B were leaking by, SI would force water into the accumulators during the monthly tests. This level change would then be detected during performance of SR 3.5.1.2. In addition, NRC approved standard OMa-1988 requires leak testing of these check valves at least once every 2 years as does the Ginna Station IST program.

Section 3.3 TS

3.3Q1 - Attachment B, CTS 2.3.1.2.g, low reactor coolant pump frequency ≥ 57.5 Hz, is marked eliminated, citing 4.v. There is no justification 4.v. Provide justification for relocating the low reactor coolant pump frequency ≥ 57.5 Hz reactor trip to the Technical Requirements Manual. (J)

Response: Justification 4.v is provided on page 168 of Attachment A. [This response was revised based on meetings week of 11/13/95. See comment #222.]

3.3Q2 - Table 3.3.1-1 in NUREG-1431 has a column labeled *Trip Setpoint* and a column labeled *Allowable Value*. Note 'a' to that table notes "Unit specific implementation may contain only *Allowable Values* depending on Setpoint Study methodology used by the unit." The Ginna ITS uses the *Trip Setpoint* column, referencing discussion 23.vii, which states the *Trip Setpoint* column is used in accordance with the setpoint methodology to reflect the licensee's nomenclature. It is noted that the submittal markup of NUREG-1431 includes both columns. NUREG-1431 allows the use of the *Allowable Value* column only, but not the *Trip Setpoint* column by itself. Provide additional bases for using only the *Trip Setpoint* column or revise ITS Table 3.3.1-1 to include the *Allowable Values*.

Response: The CTS only have Trip Setpoints specified (see CTS 2.3.1). The Allowable Values are not specified within the CTS and are instead specified within the accident analyses and the Setpoint Study program. The station calibration procedures require that the reactor trip functions be set within the Trip Setpoints specified by the TS for "as left" values (i.e., values following calibration testing). The "as found" values (e.g., trip setpoints discovered

during calibration testing), are typically allowed to be 1-2% beyond the Trip Setpoint to allow for instrument drift and testing equipment inaccuracies. Any values beyond this value result in the channel being declared inoperable. These allowances are specified in the calibration procedures and are controlled by the Setpoint Study program to ensure that the Allowable Values used in the accident analysis are always maintained. It should be noted that the 1-2% "as found" allowance is not the same as the Allowable Value used in the accident analyses. The Allowable Value used in the accident analysis includes multiple issues that go beyond drift considerations (e.g., physics issues). Consequently, there are really three types of setpoints with respect to the reactor trip system: (1) Trip Setpoints or the "as left" values which if used ensure the accident analysis setpoint assumptions to be met at all times; (2) Allowable Values which are the actual accident analysis setpoints that have no real meaning to station personnel in the field, and (3) A 1-2% value by which the Trip Setpoint can be exceeded before the accident analysis setpoint assumption is invalidated (i.e., the allowable "as found" value). Since the CTS only have a Trip Setpoint, and the instrument drift allowances are controlled by calibration procedures and the Setpoint Study program, RG&E considers that only having a Trip Setpoint column to be acceptable. In addition, Westinghouse was approached during the preparation of the May 26th submittal package concerning this issue and verbal acceptance was obtained. Westinghouse has been requested to provide additional information which is being tracked by Comment #28.

3.3Q3 - Attachment E, Page 4 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "CTS 2.3.2.2/ITS Table 3.3.1-1, 15.d". ITS Table 3.3.1-1 does not have a function 15.d. Explain the entry and justify the change.

Response: This is a typographical error in Attachment E explained as follows. For the initial internal review of LCO 3.3.1, Table 3.3.1-1 contained the reactor trip system interlocks. For that review, Function #15.d was the Power Range Neutron Flux permissive P-9. This permissive is tied into the Reactor Coolant Flow - Low, Single Loop trip as discussed in CTS 2.3.2.2 since it enables the trip function when $\geq 50\%$ RTP. Therefore, when the reactor trip system interlocks were removed from Table 3.3.1-1 following this initial review, the database which was used to generate Attachment E was not updated to remove "ITS Table 3.3.1-1, 15d" and replace it with "SR 3.3.1.16." This SR verifies that the Reactor Coolant Flow - Low, Single Loop trip is not blocked $\geq 50\%$ RTP. Comment #18 has been opened to correct this.

3.3Q4 - CTS 2.3.3.2 defines the Loss of Power and degraded voltage settings (in conjunction with the limits shown in Figure 2.3-1), "measured" values, and "acceptable" values respectively. ITS SR 3.3.4.2 appears to use defined setpoints and time delays. The Bases states the degraded voltage relays have inverse time delay characteristics. Provide justification and documentation for the Allowable Values and Trip Setpoints of ITS SR 3.3.4.2, showing the specified Allowable

Values and Trip Setpoints are adequate for the conversion to the ITS. Specifically address the acceptability of specifying the trip curve of the degraded voltage relays with a single point.

Response: The evaluation which was performed to convert the degraded voltage and loss of voltage "measured" and "acceptable" values from CTS Figure 2.3-1 to defined setpoints and time delays is documented in RG&E Design Analysis, DA-EE-93-006-08, "480 Volt Undervoltage Relay Settings and Test Acceptance Criteria." A copy of this analysis is attached. Since this analysis was used to determine the actual field setpoints and testing setpoints to ensure that CTS Figure 2.3-1 is always met, RG&E decided to put these actual values in the ITS to provide consistency.

3.3Q5 - CTS 3.12.2 limits the reactor power to 90% of rated power if the excore detectors have not had current surveillance (calibration). Corresponding ITS SR 3.3.1.6 requires calibration before reaching 90% RTP following refueling if the surveillance was not completed within 92 EFPD, but imposes no corresponding limits on reactor power. Discuss this conversion and provide justification for the lack of the requirement to limit reactor power if the calibration of the excore detectors is not current.

Response: SR 3.0.4 prevents entry into a MODE or other specified condition in the Applicability if a SR is not current. Therefore, during initial startup, SR 3.0.4 would restrict the plant from exceeding 90% RTP unless SR 3.3.1.6 is current since this Surveillance is required for the Overtemperature ΔT trip function. However, if SR 3.3.1.6 is found to be not current when $> 90\%$ RTP, SR 3.0.3 would provide up to 24 hours to perform the necessary surveillance. If SR 3.3.1.6 were not performed within the 24 hour limit, the Overtemperature ΔT trip function would be declared inoperable and LCO 3.0.3 would be entered requiring a forced shutdown to MODE 3. As such, there is a maximum 24 hour window in which the plant could be $> 90\%$ RTP with SR 3.3.1.6 not current. It is noted that CTS 3.12.2 restricts the plant to $\leq 90\%$ RTP under these conditions but there is no Completion Time for achieving this power level. Given the short time frame to complete SR 3.3.1.6 before requiring a plant shutdown, it is considered acceptable not to require a power reduction during this same time period.

3.3Q6 - CTS 4.4.7.2, a monthly calibration of the H₂ monitors is changed to a 24-month interval calibration in ITS SR 3.3.3.2, citing industry experience. Show, by historical calibration data, that the drift of the Ginna H₂ monitors supports a 24-month interval calibration cycle. Provide justification for the daily channel check (CTS 4.4.7.1) being changed to 31-days (ITS SR 3.3.3.1).

Response: This response is divided into two parts related to the calibration and channel check of the H₂ monitors. Currently, RG&E performs two types of calibration on the H₂ monitors. The first type of calibration is performed monthly and consists of turning the monitors on since they are normally in standby, and ensuring that the monitors are zeroed correctly and measuring H₂ between a span of

0% to 10% concentration. The monitors are adjusted as necessary to ensure these two parameters are acceptable. Then the monitors are supplied with two known concentrations of H₂ sample gas (5% and 9%) and all locations which indicate H₂ concentrations are then viewed to ensure that the indicated concentration is within acceptable tolerance limits. The second calibration is performed annually and consists of this same test except that all instrument strings are also calibrated against acceptable tolerance limits (i.e., "as found" values are determined during testing). The fact that the H₂ monitors are potentially adjusted with respect to their zeroed setpoint and span reading is acceptable since this is how the monitors would be used following an accident. That is, per station procedure CH-EPIP-CVH2, the monitors would be started immediately following an accident and then adjusted with respect to their zeroed setpoint and span reading. Time is allowed for these actions since as stated in the bases for LCO 3.6.7, the minimum hydrogen flammability concentration within containment would not be reached until 31 days following an accident. As such, RG&E has verified that the instrument calibrations currently performed annually would support a surveillance interval of 24 months as discussed in Attachment H of the submittal. A review of the monthly calibrations shows that between 1990 and 1994 there were only four instances where the H₂ monitor indication at either the monitor panel, remote panel, or control room meter was found out of tolerance requiring repair. In only one instance were all three monitors found out of tolerance, however, the second H₂ monitor remained OPERABLE.

The change in channel check from once daily to once every 31 days is based on NUREG-1431 which justifies the Frequency on the basis that channel failures are rare. In addition, the hydrogen monitors are only used post accident to detect high hydrogen concentration levels which could potentially lead to a breach of containment. The bases for LCO 3.6.7 state that the minimum hydrogen flammability concentration would not be reached until 31 days following an accident. This provides sufficient time to operators to detect any failure of a hydrogen monitor which would be readily observed due to the hydrogen concentration changes following the accident, and then initiate necessary repairs. Also, Ginna Station can access the Post Accident Sampling System (PASS) if required for hydrogen monitoring purposes.

3.3Q7 - WCAP-14333, May 1995, is cited as justification for placing an inoperable channel in bypass for up to 12 hours while performing routine surveillance testing of other channels. Similarly, the same document is cited as justification for allowing 72 hours to restore the channel to OPERABLE status and 6 additional hours to reduce thermal power. Describe this basis, as this May 1995 document surely has not received NRC approval.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the following will be the

changes to the CTS as addressed by Comment # 19:

- a. change the CTS limit of 2 hours for bypassing an inoperable channel to 4 hours (versus the proposed 12 hours); and
- b. change the CTS limit of 1 hour to place an inoperable channel in trip to 6 hours (versus the proposed 72 hours) before requiring a plant shutdown.

3.3Q8 - The CTS 3.6.4.2 requirement to go to hot shutdown if an inoperable hydrogen monitoring channel is not restored to operable status is changed to ITS LCO 3.3.3, Required Action C.1, to initiate preparation and submittal of a Special Report. Describe the intended Special Report, and the required time to submit it. Justify why this Special Report is an adequate action to take in lieu of hot shutdown.

Response: The bases for Required Action C.1 provide the details of what is required to be contained within the Special Report and the allowed Completion Time for submittal. Placing this information in the bases is consistent with NUREG-1431, LCO 3.3.3, Condition G. The bases also provide justification, in conjunction with change 16.ix on page 203 of Attachment A, as to why a Special Report is adequate in lieu of hot shutdown. Also, as noted in the response to 3.3Q6, the hydrogen concentration within containment is not expected to reach flammability limits until 31 days following an accident.

3.3Q9 - The CTS 3.5.3.2 requirement to go to hot shutdown if an inoperable post-accident monitoring channel is not restored to operable status is changed to ITS LCO 3.3.3, Required Action C.1, to initiate preparation and submittal of a Special Report. Describe the intended Special Report, and the required time to submit it. Justify why this Special Report is an adequate action to take in lieu of hot shutdown.

Response: The bases for Required Action C.1 provide the details of what is required to be contained within the Special Report and the allowed Completion Time for submittal. Placing this information in the bases is consistent with NUREG-1431, LCO 3.3.3, Condition G. The bases also provide justification, in conjunction with change 15.iii.c on page 199 of Attachment A, as to why a Special Report is adequate in lieu of hot shutdown.

3.3Q10 - Attachment E, Page 5 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "New/ITS Table 3.3.1-1, 15/15.i.ff." Attachment A does not appear to have a justification 15.i.ff. Explain the entry and justify the change.

Response: This is a typographical error. Therefore, this entry ("New/ITS Table 3.3.1-1, 15/15.i.ff") should be deleted from Attachment E. Comment #18 has been opened to correct this.

3.3Q11 - Attachment E, Page 5 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing

"New/ITS Table 3.3.1-1, 15/28.i.f." Attachment A, 28.i.f, appears to address channel operational tests for the power- and intermediate-range channels, but not the reactor trip breakers. Explain the entry and justify the change.

Response: This is a typographical error. Therefore, this entry ("New/ITS Table 3.3.1-1, 15/28.i.f") should be deleted from Attachment E. In its place, new entries as follows should be made:

New	ITS Table 3.3.1-1, 3	28.i.f	3.3
New	ITS Table 3.3.1-1, 4	28.i.f	3.3
New	ITS Table 3.3.1-1, 7	28.i.f	3.3
New	ITS Table 3.3.1-1, 9	28.i.f	3.3
New	ITS Table 3.3.1-1, 10	28.i.f	3.3
New	ITS Table 3.3.1-1, 11	28.i.f	3.3
New	ITS Table 3.3.1-1, 13	28.i.f	3.3

Comment #18 has been opened to correct this.

3.3Q12 - Attachment E, Page 5 of Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B, has a listing "New/ITS Table 3.3.2-1, 7/15.ii.d." The CTS markup of Table 3.5.2 includes "Add Function 7, 'ESFAS Pressurizer Pressure Interlock' 15.ii.d." Justification 15.ii.d addresses functional units 1.c and 1.d. Further, neither the NUREG-1431 markup nor the Ginna draft ITS include Function 7 in Table 3.3.2-1. Explain the entries regarding Function 7 and justify the change.

Response: This is a typographical error in both Attachment B and E. The initial internal review of LCO 3.3.2 included this function by mistake since Ginna Station does not have these ESFAS interlocks. However, this function was not removed from Attachment B and E as intended. Therefore, the entry "New/ITS Table 3.3.2-1, 7/15.ii.d" should be deleted from Attachment E while "Add Function 7, 'ESFAS Pressurizer Pressure Interlock' 15.ii.d" should be deleted from Attachment B. Comments #18 and 20 have been opened to correct these errors, respectively.

3.3Q13 - Attachment E, Page 5 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "Table 3.5-1, AS 1/ITS LCO 3.3.1/15.i.d" that is listed twice. Explain the significance of this double listing of an apparently singular item.

Response: This is a typographical error. Therefore, the second entry ("Table 3.5-1, AS 1/ITS LCO 3.3.1/15.i.d") should be deleted from Attachment E. Comment #18 has been opened to correct this.

3.3Q14 - The current Technical Specification for Table 3.5-1, Note 14, is marked "LCO 3.3.1, Conditions C, S, and U." The improved Technical Specifications, LCO 3.3.1 has no Condition U. Explain the notation, and any associated changes.

Response: This is a typographical error with respect to Condition U. Therefore, the notation in the left margin for CTS Table 3.5-1

should read "LCO 3.3.1, Conditions C and S." Comment #20 has been opened to correct this.

3.3Q15 - Attachment E, Page 6 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "Table 3.5-1, AS 14/ITS LCO 3.3.1.v," with no listing in the Notes column. Explain the significance of this listing, the change involved, and the justification for the change.

Response: This is a typographical error. Therefore, this entry ("Table 3.5-1, AS 14/ITS LCO 3.3.1.v") should be deleted from Attachment E. Comment #18 has been opened to correct this.

3.3Q16 - Justification 15.i.f states the change is discussed and justified in Reference 30. This change allows Functional Units 2 (high and low settings), 5, 6, and 7 to have an inoperable channel bypassed for up to 72 hours during surveillance testing instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4 hours, versus the current 2 hours, to bypass an inoperable channel.

3.3Q17 - Justification 15.i.g states the change is discussed and justified in Reference 30. This change allows Functional Units 2 (high and low settings), 5, 6, and 7 to have an inoperable channel bypassed for up to 12 hours during surveillance testing instead of the current 2 hours. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 4 hours, versus the current 2 hours, to bypass an inoperable channel.

3.3Q18 - ITS Table 3.3.1, Function 4, Source Range Neutron Flux, calls out Conditions I and J for Mode 2. Neither Required Action restores the inoperable channel similar to what Required Action K.1 does. Nor is a Mode reduction required to implement the Required Action (Mode 3).

How is the inoperable channel restored and what are the associated time limits?

Response: Condition I is entered if one Source Range Neutron Flux channel is inoperable while Condition J is entered if both channels are inoperable. Required Action J.1 requires that the RTBs be opened immediately which is equivalent to immediately entering MODE 3. The plant would then be in Condition L which requires the immediate suspension of positive reactivity additions. If only one channel is operable, Required Action I.1 requires the immediate suspension of positive reactivity additions. Therefore, no power increase is allowed. A power reduction is not required since the Power Range Neutron Flux - Low and Intermediate Range Neutron Flux trip functions can provide core protection. In MODES 3, 4, and 5, only the Source Range Neutron Flux provides the required core protection, consequently a power reduction is necessary if the inoperable channel is not restored.

3.3Q19 - Justification 15.i.1 states the change is discussed and justified in Reference 30. This change allows Functional Units 8, 9, 10 (low flow in one loop), 11, and 13 to have an inoperable channel bypassed for up to 72 hours during surveillance testing instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4 hours, versus the current 2 hours, to bypass an inoperable channel.

3.3Q20 - Justification 15.i.m states the change is discussed and justified in Reference 30. This change allows Functional Units 8, 9, 10 (low flow in one loop), 11, and 13 to have an inoperable channel bypassed for up to 12 hours during surveillance testing instead of the current 2 hours. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4



hours, versus the current 2 hours, to bypass an inoperable channel.

- 3.3Q21 - CTS Table 3.5-1, Action Statement 5, is marked Conditions E, N, M, & P. Justifications 15.i.l and 15.i.m note Functional Units 8, 9, 10 (low flow in one loop), 11, and 13. These become Functional Units 7.b, 8, 9.a (low flow in one loop), 13, and 12 in the ITS, with Conditions E, E, N, P, and E, respectively. ITS Table 3.3.1-1 applies Condition M to Functions 7.a (Pressurizer Pressure - Low), 9.b (Reactor Coolant Flow - Low, Two Loops), 10.b (RCP Breaker Position), and 11 (Undervoltage, Bus 11A and 11B). Provide justification showing the acceptability of Condition M for these functions.

Response: This response is organized into several parts. First, "Condition M," as placed in the left margin for CTS Table 3.5-1, Action Statement 5 is a typographical error. Comment #20 has been opened to correct this. Second, ITS Table 3.3.1-1, Function #10.b (RCP Breaker Position) was added to CTS Table 3.5-1 as part of the conversion effort (i.e., this function is currently not a technical specification requirement for Ginna Station). Since Condition M applies to this function in NUREG-1431, RG&E believes it to be acceptable (see change 15.i.w on page 193 of Attachment A). Third, Action Statement 2 to CTS Table 3.5-1 is applied to Function #7 (Pressurizer Pressure - Low) while Action Statement 6 to CTS Table 3.5-1 is applied to Function #10.b (Reactor Coolant Flow - Low, Two Loops) and #14 (Undervoltage, Bus 11A and 11B). These two Action Statements differ from Condition M of ITS in that only 1 hour (versus 6 hours) is allowed to restore an inoperable channel while an the inoperable channel is only allowed to be bypassed for 2 hours (versus 4 hours) for surveillance testing of other channels (note that RG&E is addressing the 72-hour Completion Time for restoring an inoperable channel and 12 hour bypass time separate from the conversion as stated above). The justification for increased Completion Times and bypass times between the ITS Condition M and CTS is provided in WCAP-10271-P-A which is a NRC reviewed and approved document (i.e., Reference 8 in Bases for LCO 3.3.1 as found in NUREG-1431). Comment #19 addresses the implementation of this WCAP for Ginna Station.

- 3.3Q22 - Justification 15.i.o states the change is discussed and justified in Reference 30. This change allows Functional Units 10 (low flow in both loops) and 14 to have an inoperable channel bypassed for up to 12 hours during surveillance testing instead of being tied to the next channel functional test of an operable channel. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will be revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been



opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 4 hours to bypass an inoperable channel versus being tied to the next functional test of an inoperable channel.

- 3.3Q23 - Justification 15.i.n states the change is discussed and justified in Reference 30. This change allows Functional Units 10 (low flow in both loops), 14, and 15 to have an inoperable channel bypassed for up to 72 hours during surveillance testing instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4 hours, versus the current 2 hours, to bypass an inoperable channel.

- 3.3Q24 - Discuss and justify relocating the CTS Table 3.5.1/15 requirements for 4-kVAC bus underfrequency trips.

Response: This discussion and justification is provided in change 15.i.c on page 187 of Attachment A. Please notify if additional justification is required. [This response was revised based on meeting week of 11/12/95. See comment #222.]

- 3.3Q25 - The CTS markup, Table 3.5-1, Action Statement 7, mentions 15.i.u, Condition B & C. There is no Condition C in ITS LCO 3.3.4. Explain the notation in the CTS and the lack of Condition C in ITS LCO 3.3.4.

Response: The "Condition C" in the margin for CTS Table 3.5-1, Action Statement 7 is a typographical error. The initial LCO 3.3.4 which was developed for internal review contained Conditions A, B, and C as found in NUREG-1431. However, following this internal review, Condition B was deleted and Condition C renamed Condition B as discussed in change 27.iv on page 80 of Attachment A. The markup to Action Statement 7 to CTS Table 3.5-1 was subsequently not corrected following incorporation of change 27.iv to the ITS. Comment #20 has been opened to correct this.

- 3.3Q26 - The CTS markup, Table 3.5-1, Note 5, applies to ITS Functional Units Table 3.3.1-1, 15, 16, and 17. The markup at Note 5 refers to Note (j), Functional Unit 17 (ITS). ITS Table 3.3.1-1, Functional Unit 15 (reactor trip breakers) refers to Note (k). Functional Units 16 and 17 do not refer to a footnote similar to CTS Note 5. Footnote (j) does not appear to be related to Note 5. Explain, in a correlated statement, 1) the Note 5 applicability to ITS Functional

Units Table 3.3.1-1, 15, 16, and 17, and 2) the annotated footnote (j) (CTS markup) and the Functional Unit 15 use of footnote (k) (ITS).

Response: Question 1: CTS Table 3.5-1, Note 5 only applies to ITS Table 3.3.1-1, Functions 15, 16, and 17 with respect to the "Applicable MODES or other Specified Conditions" column and associated footnote (a). There is no difference between CTS Note 5 and the required MODE of Applicability for ITS Table 3.3.1-1 Functions 15, 16, and 17. Question 2: CTS Table 3.3.1-1, Note 5 should have "FU #15, FU #16, FU #17, note (a)" in the left hand margin instead of "FU #17, note (j)" for the reasons discussed in response to Question 1. In addition, CTS Table 3.3.1-1, Note 4 should have "FU #15, Condition R, note (k)" instead of "FU #15, Condition R, note i." There is no relationship between CTS Note 5 and ITS footnote (k). Comment #20 has been opened to correct this.

3.3Q27 - Justification 15.ii.h states the change is discussed and justified in Reference 30. This change allows Table 3.3.2-1, to have an inoperable channel placed in trip within 72 hours instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #19 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip. The 6 hours is consistent with NUREG-1431 and is justified in WCAP-10271.

3.3Q28 - CTS Table 3.5.2, 3.d, *Safety Injection Start Motor Driven Pumps*, is transferred to ITS Table 3.3.2-1, 6.c, *Auxiliary Feedwater - Safety Injection*. Describe which Auxiliary Feedwater pumps are started on the Safety Injection actuation.

Response: The bases for LCO 3.7.5, *Auxiliary Feedwater System* (pages 3.7-25 and 3.7-26), describe which start signals actuate which pumps. This information is summarized below:

<u>AFW Start Signal</u>	<u>Start Motor-Driven</u>	<u>Start Turbine-Driven</u>
a. Auto logic	Yes	Yes
b. SG Level	Yes in 1/2 SGs	Yes in 2/2 SGs
c. Safety Injection	Yes	No
d. UV on Buses 11A and 11B	No	Yes
e. Trip of both MFW pumps	Yes	No

To clarify this actuation logic, RG&E would propose to either revise the bases for ITS Table 3.3.2-1, Function #6 or revise the Function

title in the table itself (e.g., Function 6.c would become Auxiliary Feedwater - Safety Injection (Motor Driven Pumps Only)). The second option is preferred. Comment #22 has been opened to address this issue.

- 3.3Q29 - Provide justification for the 48-hour restoration time instead of placing the inoperable channel in TRIP within one-hour for the auto-start of Auxiliary Feedwater on the trip of both main feedwater pumps (CTS Table 3.5-2, 3.e to ITS Table 3.3.2-1, 6.e). The ... "is justified in Reference 48" is inadequate justification to perform an evaluation of the acceptableness of the change.

Response: The bases for NUREG-1431, Actions J.1 and J.2 state that "the allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 8" which is the same as Reference 48 in Attachment A (i.e., WCAP-10271). However, upon further review, it has been determined that WCAP-10271 did not provide this justification, although it did include the allowed inoperability of 48 hours in its evaluation of other changes to ESFAS surveillance test intervals and Completion Times. The 48 hour Completion Time to restore the auto start of the motor-driven AFW pumps on trip of both main feedwater pumps also existed in previous versions of STS (see NUREG-0452). The allowance of 48 hours can be justified by the multiple actuation logic which exists for AFW. This includes the autostart of the turbine-driven AFW pump, which is redundant to both motor-driven pumps, on loss of the 4 kV buses which supply the MFW pumps. Also, the two motor-driven pumps actuate on low SG level which would occur following loss of the MFW pumps. As such, the trip of the MFW pumps is not a primary actuation function credited in the accident analysis. Comment #23 has been opened to clarify the bases justification for the 48 hour Completion Time.

- 3.3Q30 - ITS Table 3.3.2-1, Functional Unit 4.e, appears to be incorrectly tabulated, that is, the 'e.' is not in the same horizontal position as the 'a.', 'b.', 'c.', and 'd.' above it. Likewise the entry High-High Steam Flow is lined up as a subset of High Steam Flow. Correct.

Response: Comment #24 has been opened to correct this typographical error.

- 3.3Q31 - The CTS markup for Table 3.5-2, 6.b, is annotated Functional Unit 5.b and Footnote d. ITS Table 3.3.2-1 has no footnote (d). Explain the significance of the annotation and the reported footnote (d).

Response: This is a typographical error. Therefore, CTS Table 3.5-2, Functional Unit #6.b should have "FU #5.b, Footnote (c)" instead of "FU #5.b, Footnote (d)" in the left margin. In addition, CTS Table 3.5-2, Note ** at the bottom of page 3.5-12 should have "Footnote (c)" instead of "Note (d)" in the left margin. Finally, CTS Table 3.5-2, Functional Unit 5 and Note * should have "Footnote (b)" wherever "Note (c)" is used. Comment #20 has been opened to correct this.

- 3.3Q32 - Attachment E, Page 7 of 'Current Ginna TS Cross Reference to

Proposed TS - Table 2, Sorted Per Attachment B, has a listing "CTS Table 3.5-2, AS 11/ITS LCO 3.3.2/15.ii.h." Justification 15.ii.h addresses CTS Action Statement 12, Functional Unit 3.c, not Action Statement 11 as listed. Explain the entry, resolve the discrepancy, and justify the change.

Response: This is a typographical error. Therefore, this entry ("Table 3.5-2, AS 11/ITS LCO 3.3.2/15.ii.h") should be deleted from Attachment E. Comment #18 has been opened to correct this.

3.3Q33 - Justification 15.ii.i and 15.ii.j state the changes are discussed and justified in Reference 30. This change allows Table 3.3.2-1, Functional Unit 2.b, to place an inoperable channel in bypass for 12 hours while performing surveillance testing on a redundant channel and to have an inoperable channel placed in trip within 72 hours instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #21 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4 hours, versus the current 2 hours, to bypass an inoperable channel.

3.3Q34 - Justification 15.ii.h states the change is discussed and justified in Reference 30. This change allows Table 3.3.2-1, Functional Unit 3.c, to have an inoperable channel placed in trip within 72 hours instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: See response to 3.3Q27 (same question).

3.3Q35 - Justification 15.ii.m states the change is discussed and justified in Reference 30. This change allows Table 3.3.2-1, Functional Units 3.b.ii, 5.a and 5.b, to have an inoperable channel placed in trip within 72 hours instead of the current 1 hour. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation

of TOPS (i.e., WCAP-10271). Comment #21 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip. The 6 hours is consistent with NUREG-1431 and is justified in WCAP-10271.

3.3Q36 - Attachment E, Page 7 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "CTS Table 3.5-2, AS 6/ITS LCO 3.3.2/15.ii.o." Justification 15.ii.o addresses CTS Action Statement 2, Functional Unit 3.f, not Action Statement 6 as listed. Explain the entry, resolve the discrepancy, and justify the change.

Response: This is a typographical error. The entry "Table 3.5-2, AS 6/ITS LCO 3.3.2/15.ii.o" was most likely intended to be "Table 3.5-2, AS 6/ITS LCO 3.3.2/15.i.o" in Attachment E. However, this entry would also be incorrect since the change documented in 15.i.o is only applicable to the use of CTS Action Statement 6 in ITS Table 3.3.1-1 and not ITS Table 3.3.2-1. Comment #18 has been opened to delete this entry in Attachment E.

3.3Q37 - Justification 15.ii.e and 15.ii.f state the changes are discussed and justified in Reference 30. This change allows Table 3.5-2, Functional Units 1.c, 1.e, 1.d, 6.b, 4.c and 5.b, to place an inoperable channel in bypass for 12 hours while performing surveillance testing on a redundant channel and to have an inoperable channel placed in trip within 72 hours instead of the current 1 hour. Reference 30, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271). Comment #21 has been opened to address this. However, to support this review, the CTS will be changed for the identified functional units to allow 6 hours, versus the current 1 hour, to place the inoperable channel in trip and to allow 4 hours (or 8 hours as allowed by the WCAP), versus the current 2 hours, to bypass an inoperable channel.

3.3Q38 - The following values in ITS Table 3.3.2-1 come from CTS Table 3.5-4, with changes as noted

ITS Function	CTS Value	ITS Value	Justification
1.c, Allowable Value	≤ 5.0 psig	≤ 6.0 psig	15.ii.q
1.d, Setpoint	≥ 1723 psig	≥ 1750 psig	--
1.e, Allowable Value	≥ 500 psig	≥ 358 psig	15.ii.q
2.c, Allowable Value	≤ 40 psig	≤ 32.5 psig	15.ii.q

ITS Function	CTS Value	ITS Value	Justification
4.d, Setpoint	$\leq 0.4E6$ lbm/hr @ 755 psig, $\leq 545^\circ\text{F}$	dp corresponding to $\leq 0.49 \times 10^6$ lbs/hr at 755 psig, $T_{\text{avg}} \leq 545^\circ\text{F}$	--
4.d, Allowable Value	$\leq 0.55E6$ lbm/hr @ 755 psig, $\leq 545^\circ\text{F}$	dp corresponding to $\leq 0.55 \times 10^6$ lbs/hr at 755 psig, $T_{\text{avg}} \leq 545^\circ\text{F}$	--
4.e, Allowable Value	$\leq 3.7E6$ lbm/hr @ 755 psig	dp corresponding to $\leq 3.7 \times 10^6$ lbs/hr at 755 psig	--

Justification 15.ii.q states the revision reflects the accident analysis. Provide justification showing the setpoints and allowable values are in accordance with the Ginna Setpoint Analysis.

Response: Please see table below:

ITS Table 3.3.2-1				Setpoint Analysis Assumptions		
Function #	Function Description	Trip Setpoint	Allowable Value	Trip Setpoint	Allowable Value	RG&E Calculation #
1.c	Safety Injection - Containment Pressure - High	≤ 4.0 psig	≤ 6.0 psig	4.0 psig	≤ 5.0 psig	DA-EE-92-041-21
1.d	Safety Injection - Pressurizer Pressure - Low	≥ 1750 psig	≥ 1715 psig	1750 psig	≥ 1711 psig	DA-EE-92-087-21
1.e	Safety Injection - Steam Line Pressure - Low	≥ 514 psig	≥ 358 psig	514 psig	≥ 500 psig	DA-EE-92-088-21
2.c	Containment Spray - Containment Pressure - High High	≤ 28 psig	≤ 32.5 psig	28 psig	≤ 30 psig	DA-EE-92-041-21
4.d	Steam Line Isolation - High Steam Flow	≤ 0.4E6 lbm/hr @ 755 psig	≤ 0.55E6 lbm/hr @ 755 psig	0.49E6 lbm/hr @ 755 psig	≤ 0.55E6 lbm/hr @ 755 psig	DA-EE-92-089-21
4.e	Steam Line Isolation - High High Steam Flow With SI	≤ 3.6E6 lbm/hr @ 755 psig	≤ 3.7E6 lbm/hr @ 755 psig	3.6.E6 lbm/hr @ 755 psig	≤ 3.7E6 lbm/hr @ 755 psig	DA-EE-92-089-21

As can be seen from this table, the trip setpoints as used in the Setpoint Analysis are the same as those contained in ITS Table 3.3.2-1 (i.e., the trip setpoints in the Setpoint Analysis are the maximum or minimum trip setpoint specified in the ITS table). The only exception is Function 4.d where the Setpoint Analysis assumed a higher trip setpoint than the ITS (i.e., a trip setpoint closer to the Allowable Value). However, a higher trip setpoint is conservative in this instance since the Allowable Value remains the same in both the Setpoint Analysis and ITS Table. In addition, the Allowable Value as assumed in the Setpoint Analysis is either equivalent or more limiting than that specified in the ITS table such that the Setpoint Analysis remains bounding. The only exception is with respect to Function 1.d in which the Allowable Value in ITS is more limiting than that used in the Setpoint Analysis. Therefore, RG&E will revise DA-EE-92-087-21 to reflect

this change. Comment #29 has been opened to address this.

3.3Q39 - The Allowable Values and Trip Setpoints for SR 3.3.4.2 were derived from CTS Figure 2.3-1, which was not included in the 3.3 or 3.8 tabs for review. The ITS Bases B 3.3.4, Reference 3, may clarify the transition. Verify that the Allowable Values and Trip Setpoints for SR 3.3.4.2 are correctly incorporated into the ITS from CTS Figure 2.3-1.

Response: Figure 2.3-1 is on page 2.3-10 of the CTS as included within the 3.3 tab of Attachment B. Please see response to question 3.3Q4 (same question).

3.3Q40 - Attachment E, Page 7 of 'Current Ginna' TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "CTS Table 3.5-4, 8.a/ITS Table 3.3.2-1, 7." There is no Functional Unit 7 associated with ITS Table 3.3.2-1. Explain the entry, resolve the discrepancy, and justify the change.

Response: This is a typographical error. The entry "CTS Table 3.5-4, 8a/ITS Table 3.3.2-1, 7" should be replaced with "CTS Table 3.5-4, 8a/ITS Table 3.3.2-1, Footnote (a)" and "CTS Table 3.5-4, 8a/ITS SR 3.3.2.6." Comment #18 has been opened to correct this. In addition, the CTS markup for Table 3.5-4, 8a should be revised to replace "FU #7" with "Footnote (a)" and "SR 3.3.2.6" in the left margin. The basis for these changes is described in ITS Change 24.ii on page 75 of Attachment A. Comment #20 has been opened to correct this.

3.3Q41 - Justification 28.i.a states 'various calibration and testing interval requirements for RTS and ESFAS Functions were revised consistent with NUREG-1431.' Consistency with the NUREG does not make the change of interval acceptable, unless factors such as instrument stability, lack of instrument drift, and setpoint calculations document the acceptability of the extended interval. What are the bases for the acceptability of the calibration and testing interval extensions?

Response: These changes in calibration and testing intervals are being addressed by a separate submittal related to incorporation of NRC approved WCAP-10271 (i.e., TOPS). Comments #19 and #21 have been opened to track this issue.

3.3Q42 - The change in the channel operational test from monthly to quarterly for the following have not (apparently) been justified:

- a. steam generator water level - high (CTS Table 4.1-1, Function 11/ITS Table 3.3.2-1, Functional Unit 5.b, SR 3.3.2.2)
- b. steam generator water level - low-low (ITS Table 3.3.2-1, Functional Unit 6.b, SR 3.3.2.2)
- c. steam generator water level (narrow range) (ITS Table 3.3.3-1, Functional Unit 20, no Surveillance Requirement)
- d. reactor containment pressure (CTS Table 4.1-1, Functional Unit 17/ITS Table 3.3.2-1, Functional Units 1.c and 2.c, SR

- 3.3.2.2).
- e. intermediate range neutron flux - high (CTS Table 4.1-1, Functional Unit 2/ITS Table 3.3.1-1, Functional Unit 3, SR 3.3.1.8)
 - f. steam generator pressure - low (CTS Table 4.1-1, Functional Unit 26/ITS Table 3.3.2-1, Functional Unit 1.e, SR 3.3.2.2)
 - g. steam flow - high (CTS Table 4.1-1, Functional Unit 32/ITS Table 3.3.2-1, Functional Unit 4.d, SR 3.3.2.2)
 - h. steam flow - high-high (CTS Table 4.1-1, Functional Unit 32/ITS Table 3.3.2-1, Functional Unit 4.e, SR 3.3.2.2)
 - i. T_{avg} - low (CTS Table 4.1-1, Functional Unit 33/ITS Table 3.3.2-1, Functional Unit 4.d, SR 3.3.2.2)
 - j. control room air intake radiation detectors (CTS Table 4.1-1, Functional Unit 36/ITS SR 3.3.5.1)
 - k. overtemperature ΔT (CTS Table 4.1-1, Functional Unit 4/ITS Table 3.3.1-1, Functional Unit 5, SR 3.3.1.7)
 - l. overpower ΔT (CTS Table 4.1-1, Functional Unit 4/ITS Table 3.3.1-1, Functional Unit 1, SR 3.3.1.7)
 - m. reactor coolant flow (CTS Table 4.1-1, Functional Unit 5/ITS Table 3.3.1-1, Functional Unit 9, SR 3.3.1.7)
 - n. pressurizer water level - high (CTS Table 4.1-1, Functional Unit 6/ITS Table 3.3.1-1, Functional Unit 8, SR 3.3.1.7)
 - o. pressurizer pressure - low/high (CTS Table 4.1-1, Functional Unit 7/ITS Table 3.3.1-1, Functional Unit 7, SR 3.3.1.7)
 - p. pressurizer pressure - low (CTS Table 4.1-1, Functional Unit 7/ITS Table 3.3.2-1, Functional Unit 1.d, SR 3.3.2.2)
 - q. 4-kVac undervoltage/underfrequency - Buses 11A and 11B (CTS Table 4.1-1, Functional Unit 8/ITS Table 3.3.1-1, Functional Unit 11, SR 3.3.1.9)
 - r. 4-kVac undervoltage/underfrequency - Buses 11A and 11B (CTS Table 4.1-1, Functional Unit 8/ITS Table 3.3.2-1, Functional Unit 6.e, SR 3.3.2.3)

Provide those justifications.

Response: *These changes in calibration and testing intervals are being addressed by a separate submittal related to incorporation of NRC approved WCAP-10271 (i.e., TOPS). Comments #19 and #21 have been opened to track this issue.*

3.3Q43 - Attachment E, Page 8 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "CTS Table 4.1-1, 21/ITS Table 3.3.1-1, 4." Describe how the valve temperature interlocks (CTS Table 4.1-1, Functional Unit 21) relate to the source range neutron flux instrument channels (ITS Table 3.3.1-1, Functional Unit 4). It appears to be a discrepancy. Explain the entry, resolve the discrepancy, and justify the change.

Response: *This is a typographical error. CTS Table 4.1-1 Function 21 (valve-temperature interlocks) refers to a refueling basis test of MSIV isolation (note- this Function was added as part of post-TMI technical specification changes which were required to ensure that all ESFAS were being tested). Therefore, the entry in Attachment E should be "CTS Table 4.1-1, 21/SR 3.3.2.4" which is a refueling*

basis TADOT. The only difference between CTS and ITS is the refueling interval which is discussed and justified in Attachment H of the May 26th submittal. Change #18 has been opened to correct Attachment E.

3.3Q44 - Attachment E, Page 8 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has the following listings:

CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 1
CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 2
CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 3
CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 4
CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 5
CTS Table 4.1-1, 22/ITS Table 3.3.2-1, 6.

What is the relation between the pump-valve interlock (Functional Unit 22 of CTS Table 4.1-1) and the listed ESFAS actuation system instrumentation as noted? How is the refueling interval channel check of the pump-valve interlock implemented in the ITS? Explain the entries, resolve the discrepancies, and justify the changes.

Response: CTS Table 4.1-1, Functional Unit 22 (Pump-Valve Interlock) requires a check each refueling outage that all pumps and valves which receive an ESF Signal actuate as required. Therefore, this requirement is related to all functions listed in ITS Table 3.3.2-1 (i.e., Functions 1-6) and is implemented by the TADOT specified for each Automatic Actuation Logic and Actuation Relays subfunction specified in the table (i.e., SR 3.3.2.4). These TADOTs are required every 24 months which is the new refueling interval. Therefore, except for the refueling interval issue which is addressed in Attachment H of the submittal, there is no technical change in converting this CTS requirement.

3.3Q45 - Attachment E, Page 8 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' lists "CTS Table 4.1-1, 23/ITS Table 3.3.1-1, 14 (Turbine Trip Setpoint - Safety Injection input from ESFAS - Calibration and test interval). What is the relation between the turbine trip setpoint and the Block Trip? How are the calibration and test intervals implemented in the ITS? Explain the entries, resolve any discrepancies, and justify the changes. Justify relocation of the block trip note to UFSAR and Bases.

Response: This response is organized into several parts. First, the Attachment E listing is a typographical error. This entry should read "CTS Table 4.1-1, 23/ITS Table 3.3.1-1, 13.a" (Turbine Trip - Low Autostop Oil Pressure). Comment #18 has been opened to address this. In addition, the Attachment B markup of CTS Table 4.1-1, Function 23 should be revised to replace "FU #14, (3.3.1)" in the left margin with "FU #13.a, (3.3.1)." Comment #20 has been opened to correct this. Second, the monthly "Block Trip" test identified in CTS Table 4.1-1, 23 is currently performed by closing a stop valve and verifying that one of the three channels trips. The "Block Trip" note refers to performing a logic combination test.

This cannot be performed at power since it would cause a reactor trip, hence, the trip is blocked by not performing the logic combination test. Instead, this is performed as part of the refueling outage basis calibration. Third, the conversion of these turbine trip function refueling outage calibration and monthly test to the ITS is as follows. The calibration of Low Autostop Oil Pressure Turbine Trip Setpoint will be performed each refueling outage per SR 3.3.1.10 which is consistent with CTS Table 4.1-1, 23 (note - the refueling outage interval has increased from 18 months to 24 months as discussed and justified in Attachment H to the May 26th submittal). The monthly test will now be replaced with a TADOT (SR 3.3.1.12) performed once every reactor startup if it has not been performed within the last 31 days. This change in surveillance interval is justified in Attachment H to the May 26th submittal. Fourth, the relocation of the "Block Trip" note from CTS Table 4.1-1 is acceptable since with the incorporation of SR 3.3.1.12, this note is no longer required (i.e., the test will no longer be performed in a condition which could potentially result in a reactor trip).

3.3Q46 - CTS Table 4.1-1, Functional Unit 40, Manual Trip Breaker, is tested on a refueling basis. The 'R' is circled on the markup, with a '12' attached. ITS Table 3.3.1-1, Functional Unit 1, requires a Trip Actuation Device Operational Test every 24 months. Resolve the discrepancy between the noted 12 months and the 24 month interval. Justify the change from refueling to 24 months.

Response: The '12' provided in the marked up CTS Table 4.1-1, Functional Unit 40, refers to SR 3.3.1.12. That is, the "Check", "Calibrate", and "Test" columns for CTS Table 4.1-1 are identified by the applicable LCO # and the associated SR #. For example, for Functional Unit 40, the marked up text in the "Test" column shows "(3.3.1) - 12." This actually means, LCO 3.3.1, SR 3.3.1.12. However, this is a typographical error in that a '11' should be provided in the "Test" column since SR 3.3.1.11 is the actual surveillance. In addition, the '12' in the "Test" column for Functional Unit 41.b should also be "11." Comment #20 has been opened to correct this. The justification for the change from refueling (or 18 months) to 24 months is provided in Attachment H of the May 26th submittal.

3.3Q47 - Attachment E, Page 8 of 'Current Ginnà TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' lists "CTS Table 4.1-1, 8/ITS Table 3.3.1-1, 11/28.i.c" (4-kV voltage and frequency). What is the disposition of the underfrequency relays and the justification for that action?

Response: The 4 kV underfrequency reactor trip function has been relocated to the TRM as discussed in change 15.i.c on page 187 of Attachment A. Therefore, the CTS markup of Table 4.1-1, Function 8 should be revised from "FU #11 & 12 (3.3.1)" to "FU #11 (3.3.1)". This error was created during the initial draft of LCO 3.3.1 which included the 4 kV underfrequency reactor trip function (i.e., FU #12) that was subsequently relocated. In addition, "15.i.c" should be added to the left margin of the CTS markup. Comment #20 has been opened to correct this. Comment #18 has also been opened to add "CTS Table 4.1-1, 8/ITS Table 3.3.1-1, 11/15.i.c" to Attachment E.



3.3Q48 - Attachment E, Page 8 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' lists "CTS Table 4.1-1, 8/ITS Table 3.3.1-1, 12/28.i.c" (4-kV voltage and frequency in the CTS and Steam Generator (SG) Water Level Low-Low in the ITS). What is the relation of the two? What change is taking place here? Justify that change.

Response: As discussed above in the response to 3.3Q47, this entry is a typographical error. Therefore, the entry "CTS Table 4.1-1, 8/ITS Table 3.3.1-1, 12/28.i.c" should be deleted from Attachment E. Comment #18 has been opened to correct this.

3.3Q49 - The ITS revises the NUREG-1431 completion time to place an inoperable channel in TRIP from 6 hours to 72 hours, and the time limit an Inoperable channel may be bypassed for Surveillance Testing from 4 hours to 12 hours. Justification C.23.i states this is justified in Reference 30. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271) which is what the bypass and Completion Times of NUREG-1431 are based upon. Comment #19 has been opened to address this.

3.3Q50 - Justification C.23.i states the frequencies of SR 3.3.1.3 and SR 3.3.1.6 (for the overtemperature ΔT) were "revised consistent with Ginna Station practices." Where are these frequencies located in the current Technical Specifications or other documentation? If included in the current Technical Specifications, is there a change in the frequency and, if so, what is the basis for the change in the frequency? Note, the NUREG markup does not include SR 3.3.1.3 for overtemperature ΔT while the ITS correctly does.

Response: SR 3.3.1.3 is not required within the CTS while SR 3.3.1.6 is only partially addressed by CTS 3.12.1. However, none of the changes made to SR 3.3.1.6 impact the requirements of CTS 3.12.1. Ginna Station currently performs these two surveillances in the form described by the revised surveillance requirements and their associated bases. These surveillances are implemented and controlled by station procedures. Any changes to these SRs as provided in the May 26th submittal could have a significant impact on station operations with no corresponding benefit. The missing SR 3.3.1.3 in the NUREG markup is a typographical error. Comment #20 has been opened to correct this.

3.3Q51 - The ITS revises the NUREG-1431 trip setpoints to plant specific values. Justification C.23.xxviii states "the Trip Setpoints values for various trip functions was replaced with a note stating these values are 'based on established limits'." Further, these trip

setpoints are controlled within plant procedures and the setpoint methodology program. This applies to the source- and intermediate-range neutron flux instrumentation, the Buses 11A and 11B undervoltage instrumentation, and low autostop oil pressure turbine trip. For the record, where are these setpoints located and how are they controlled. Describe the acceptability of this system of setpoint control.

Response: The setpoints for the Intermediate Range Neutron Flux trip function, Source Range Neutron trip function, Undervoltage Bus 11A and 11B trip function, and Turbine Trip - Low Autostop Oil Pressure trip function were not added to the ITS since these setpoints are not in our CTS. In addition, these trip functions are backup functions not specifically credited in the accident analysis (i.e., these functions are credited as a backup for conservatism and uncertainty considerations but are not modeled within the analyses using an actual trip setpoint). The setpoints for these trip functions are contained within numerous documents within RG&E including station procedures (e.g., calibration procedures, setpoint procedures), the UFSAR (Section 7.2.2.2), and the Setpoint Study program. The affected procedures and the UFSAR are controlled under 10 CFR 50.59 while the Setpoint Study program would have to be notified of a setpoint change. RG&E considers this to provide sufficient control.

3.3Q52 - Expand on justification C.24.xv. What in Reference 48 supports the extension of the channel operational test from monthly to quarterly? The statement is made that the ESFAS design does not allow testing of the Actuation Logic, the Master Relays, or the Slave Relays monthly. What testing assures their Operability and where are these requirements located? Justify not having those requirements in the ITS.

Response: Reference 48 (i.e., WCAP-10271) generically supports an increase in the channel operational test from monthly to quarterly (see NRC letter to G.T. Goering, WOG, dated April 30, 1990). The plant-specific evaluation of the historical test data necessary to support this change in testing frequency is to be provided separately by RG&E per Comment #21. The ESFAS design which allows monthly testing of the actuation logic, master relays, and slave relays includes the use of installed bypass capability and low voltage signals which are insufficient to pick up a relay but capable of verifying circuit continuity (see NUREG-1431 bases for SR 3.3.2.2 through 3.3.2.4). This design is used by solid state protection systems (e.g., EAGLE 21), but not by relay driven systems as installed at Ginna Station. The only method of verifying the actuation logic, master relays, and slave relays at Ginna Station is to actually inject a signal to physically actuate the end device components (i.e. perform a TADOT). This test can only be performed while shutdown since a reactor trip would otherwise occur. Performance of a refueling outage interval TADOT (SR 3.3.2.4) is consistent with the CTS requirements (Table 4.1-1, Function #22). Therefore, current testing requirements are retained within ITS. The other Westinghouse 2-Loop plants which are of similar design have implemented WCAP-10271 without revising this testing frequency.

3.3Q53 - The ITS revises the NUREG-1431 completion time to place an inoperable channel in TRIP from 6 hours to 72 hours, and the time limit an Inoperable channel may be bypassed for Surveillance Testing from 4 hours to 12 hours. Justification C.24.i states this is justified in Reference 30. Reference 30, *Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times*, May 1995, has not been reviewed and approved. Further, the Sections applicable to the present justification are not available to this reviewer. Provide stand-alone justification for this change.

Response: RG&E has agreed to address the issues justified in WCAP-14333 separate from the conversion. Therefore, RG&E will revise LCO 3.3.1 and 3.3.2 to remove all of these issues and request implementation of TOPS (i.e., WCAP-10271) which is what the bypass times and Completion Times in NUREG-1431 are based upon. Comment #21 has been opened to address this.

3.3Q54 - ITS Table 3.3.3-1 specifies '2' Steam Generator Water Level (Narrow-range) channels and '2 per SG' Steam Generator Water Level (Wide-range) channels. As all the listed instrumentation is either Type A variables or Category 1 instrumentation, shouldn't the number of required channels listing for the narrow-range channels also read '2 per SG'?

Response: Yes, Table 3.3.3-1 should be revised for the Steam Generator Water Level (Narrow-range) to include 'per SG' in the required channel column consistent with the wide range requirement. Comment #25 has been opened to correct this.

3.3Q55 - Describe the loss of power/degraded voltage detection, logic, and actuation for diesel generator starting. A drawing would be helpful. The NUREG markup and the ITS state that there need to be two OPERABLE channels per 480-Vac safeguards bus. Justification C.27.iv states the logic uses "one-out-of-two logic taken twice" and that "both channels must trip to operate a LOP DG start." Clarify these meanings.

Response: The requested drawing is attached. As can be seen from this drawing, each required channel is actually comprised of two relays: a degraded voltage relay and a loss of voltage relay. One relay from each channel must actuate in order to generate an undervoltage signal on the bus. RG&E proposes to add this drawing to the bases for LCO 3.3.4 with additional supporting bases text. Comment #26 has been opened to address this.

3.3Q56 - Justification C.29.iii states the Completion Time of 48 hours for Condition A for the CREATS is discussed in justification (D.)15.vii. While D.15.vii discusses the CREATS, it does not discuss the Completion Time of 48 hours for Condition A. Augment justification C.29.iii to stand alone and provide the justification for the 48 hour Completion Time.

Response: The discussion in change C.29.iii is a typographical error that was created during the development of the initial LCO 3.3.5 for internal

review. The 48 hour Completion Time discussed in change C.29.iii only pertains to Condition A of LCO 3.7.9 which is consistent with CTS requirements. Condition A of LCO 3.3.5 only allows 1 hour to restore an inoperable channel consistent with CTS 3.5.6.2. Comment #27 has been opened to delete the entire sentence containing this discussion in change C.29.iii. In addition, RG&E noticed that the marked-up NUREG does not revise the Completion Time for LCO 3.3.7, Condition A from 7 days to 1 hour as provided in the ITS. Comment #24 has been opened to correct this.

3.3Q57 - Attachment E, Page 5 of 'Current Ginna TS Cross Reference to Proposed TS - Table 2, Sorted Per Attachment B,' has a listing "New/ITS Table 3.3.1-1, 14/15.i.x." Attachment A does not appear to have a justification 15.i.x. Explain the entry and justify the change.

Response: Change 15.i.x is provided on page 193 of Attachment A.

3.3Q38A Current Technical Specifications Table 3.5-4, Functional Unit 2.b, CONTAINMENT SPRAY, High-High Containment Pressure, is moved to the improved Technical Specifications Table 3.3.2-1, Functional Unit 2.c, Containment Spray, Containment Pressure - High. The allowable value was ≤ 40 psig and is ≤ 32.5 psig in the improved Technical Specifications. The setpoint is the same in both versions of the Technical Specifications. The allowable value as revised is less conservative than the value in the accident analyses, calculation DA-EE-92-041-21, ≤ 30 psig. Justify using ≤ 32.5 psig as the allowable value in the improved Technical Specifications.

Response: This is a typographical error in Attachment A, Section C, item 94.vi in that ITS SR 3.8.1.6 should not have a note restricting performance of this SR in MODES 3 and 4. This SR is the verification that the alternate circuit distribution network to the shutdown loads is OPERABLE. The NUREG-1431 bases state that this SR should not be performed with the reactor critical since performance of the SR could cause perturbations to the electrical distribution systems. Therefore, a Note is applied which restricts performance of this test during MODES 1 and 2. RG&E has proposed to remove this Note for the following reasons. The offsite circuitry for Ginna Station is designed such that the two offsite power sources can each supply its respective safeguards train (50/50 mode) or that one of the two offsite power sources can supply both safeguards trains (100/0 or 0/100 mode). The plant is normally run in the 50/50 mode but does change to the 100/0 or 0/100 mode depending on maintenance activities and weather related issues. That is, one of the two offsite power sources is more vulnerable to lightening strikes and, as such, during severe weather conditions is removed from service and the plant realigned to the 100/0 mode. Realigning the offsite power sources accomplishes ITS SR 3.8.1.6. Since the plant normally performs these activities at power without any electrical distribution system perturbations, restricting this activity in MODES 1, 2, 3, or 4 is unnecessary and unwarranted. Comment #33 has been opened to remove reference to ITS SR 3.8.1.6 from item 94.vi.

3.3Q6A The response to 3.3.Q6 states a review of the monthly calibrations

for the H₂ monitor shows that, between 1990 and 1994, there were only four instances where the H₂ monitor indication at either the monitor panel, remote panel, or control room meter was found out of tolerance and requiring repair. This is a failure rate of approximately 3% (four failures out of approximately 150 calibrations). In only one instance, the response continues, were all three monitors out of tolerance, however, the second H₂ monitor remained Operable. Explain the later statement and describe the acceptability of the failure rate.

Response: The event in which the three monitors were out of tolerance but the second monitor remained OPERABLE relates to the fact that all three monitors were declared inoperable per the calibration procedure. However, the second monitor remained OPERABLE, it just had to be adjusted. As such, even in this instance, there was a means of determining hydrogen concentrations. The failure rate is considered acceptable since in no case did the monitor actually fail, it only provided readings which were not within established tolerances. Therefore, the monitors would still provide some form of indication.

Section 3.6 Improved TS

57. ITS 3.6.1

- i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers associated with subatmospheric, ice condenser and dual containment structures were deleted. This is an ITS Category (iv) change. ACCEPTABLE
- ii. The Note in the Frequency column for SR 3.6.1.1 was moved to the Surveillance column as preferred by licensed personnel. In addition, "containment mini-purge valve" was added to the text of SR 3.6.1.1 as an exemption since the mini-purge valve leakage acceptance criteria is specified in new SR 3.6.3.4. These are ITS Category (iv) changes. Also, approved Traveller BWR-14, C.1 was only partially incorporated due to the proposed new Appendix J rule which was recently published for comment (Ref. 22). The changes provide consistency with the proposed new rule.

[ITS57ii]:
3.6Q1

The "Note" change is rejected as not conforming to ITS format. This may be an ITS generic item in all chapters worthy of a generic solution.

Status: []
Response:

Rejected

Based on a telephone conversation with Carl Schulten on 8/23/95, the NRC has agreed that the relocation of Surveillance Frequency notes is a "good practice" and that either the NRC or the industry will generate a Traveller to incorporate. Comment #34 has been opened to track this resolution. Suggest change status to "open" for the interim.

3.6Q2

How are these valves different from all other CIVs? If so, shouldn't we add in the shutdown purge valves and all other CIVs

that have different testing requirements? Are they tested during Integrated Leak Rate Type A testing? Are they Type A tested separately?

Status: Open

Discussion: *The mini-purge valves are different from all other valves in that a specific leakage acceptance limit is specified in ITS SR 3.6.3.4 ($\leq 0.05 L_a$). Other than airlocks and the mini-purge valves, the ITS do not have any other containment barriers with "special" leakage limits. Currently, NUREG-1431 has a exemption in SR 3.6.1.1 with respect to airlocks. The bases state that this exemption is added since "failure to meet air lock [and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to Type A, B, and C leakage causes that to exceed these limits." RG&E is proposing to retain the bases text in []s for the mini-purge valves and add specific reference to the mini-purge valves in SR 3.6.1.1 similar to the airlocks. Without this text in SR 3.6.1.1, the failure to meet ITS SR 3.6.3.4 would imply that containment is inoperable which is not accurate.*

3.6Q3 In SR 3.6.1.1, second paragraph is relocated to definition of L_a per traveler but is also replaced by new paragraph from NUREG-1431 Rev.1 as follows:

The leakage rate acceptance criterion is $\leq 1.0 L_a$. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are $< 0.6 L_a$ for the Type B and Type C tests, and $< 0.75 L_a$ for the Type A test.

Status: Open

Discussion: *This text has been relocated to the bases for LCO 3.6.1.1 since it is only an expansion of containment OPERABILITY as provided in Appendix J. Also, this text is removed in support of the new Appendix J rule (see response to 3.6Q4).*

3.6Q4 Incorporating changes based upon anticipated rule change just noticed in the Federal Register is outside conversion to the STS. What is in Reference 22?

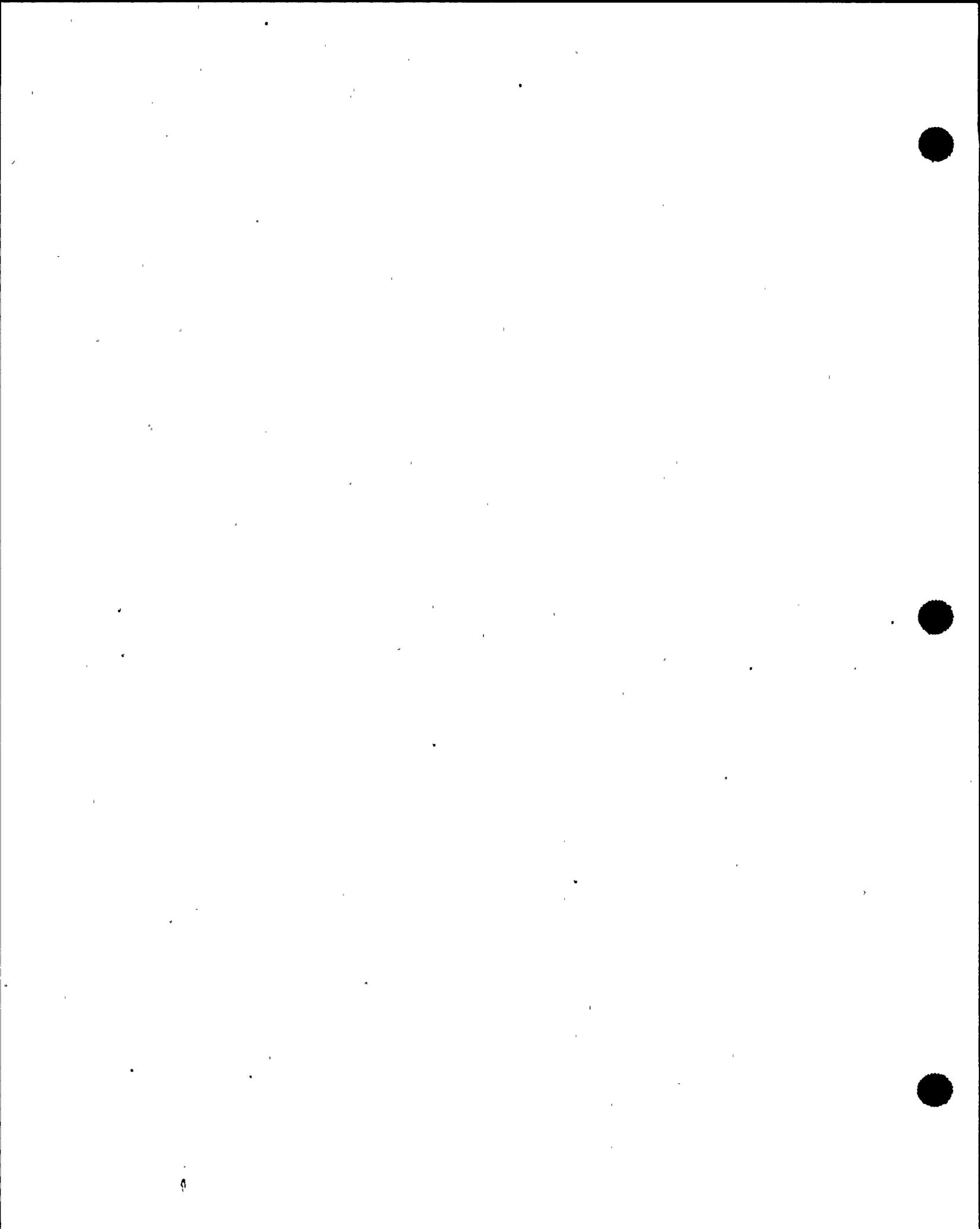
Status: Rejected

Discussion: *The new Appendix J rule technical specification changes are currently being prepared by the industry and NRC with a traveller expected in mid-September (Note letter from C. Grimes to Owner's Groups dated July 28, 1995). Comment #35 has been opened to track this issue. Suggest change status to "open" in the interim.*

iii. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added with respect to containment.

b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.



[ITS57.iii.a and b]:

3.6Q5 There are many BASES references to exemptions, AIF GDC criteria and staff evaluations provided which can not be confirmed. Also, shouldn't the offsite dose limit be consistently defined as "well within" or "a fraction of" the limit rather than just "within" the limit?

Status: Open

Discussion: *There are two statements with respect to 10 CFR 100 limits in the LCO 3.6.1 bases. The first bases statement is with respect to maintaining the containment OPERABLE since this limits leakage to the outside environment to within the Part 100 limits. The second statement is that an OPERABLE containment, in combination with the minimum safeguards equipment results in doses well within the Part 100 limits. These are two different issues since the first statement provides the reason for maintaining an OPERABLE containment (i.e., to ensure that offsite doses are within Part 100 limits) while the second statement implies is that if containment and the associated safeguards equipment are OPERABLE, then the calculated offsite doses are well within these Part 100 limits. Therefore, RG&E does not believe a bases change is necessary.*

3.6Q6 Inserts 3.6.1.4 and 5 do not match the NUREG-1431 Rev #1. Explain this BASES change regarding minimum and maximum pathway leakage rates?

Status: Open

Discussion: *Inserts 3.6.1.4 and 3.6.1.5 are based on the new Appendix J rule and how to actually interpret the "missing" text discussed in 3.6Q3 above. The definition of the minimum and maximum pathway leakage is provided in ANSI Std 56.8 and is summarized below. Minimum pathway leakage is the leakage that is assigned from the smallest leakage of two in-series valves. Maximum pathway leakage is the leakage that is assigned from the largest leakage of two in-series valves. The addition of this text and use of these terms is in fact, a more restrictive change since the Type B and C leakage must be < 0.6 L_g using the maximum pathway leakage immediately following plant startup and < 0.6 L_g using the minimum pathway leakage at all other times. Hence, margin is built into the Type B and C leakage tests to allow increased leakage following plant startup to ensure that the overall leakage limit of 0.6L_g is not exceeded. Since this is directly tied to the new Appendix J rule, comment #35 has been opened to track this issue.*

3.6Q7 The use of isolation barriers has been requested many places in lieu of containment isolation valves. This is dealt specifically in LCO 3.6.3. The proposed text added here is rejected.

Status: Rejected

Discussion: *To be discussed at the meeting.*

3.6Q8 Changes to Applicability suggest there is no containment operability during MODE 6, Refueling is implied as not applicable.

Status: Open

Discussion: *RG&E has proposed to relocate the MODE 6 containment requirements to the TRM as discussed in Attachment A, Section C, item 107.i. [This response was revised per comment #221]*

3.6Q9 See ITS57.iv below.
Status: Open
Discussion: See responses to ITS 57.iv below (i.e., 3.6Q11 and 3.6Q12).

3.6Q10 Not used.

iv. Incorporation of approved Traveller BWR-15, C.8. Not checked

[ITS57.iv]:

3.6Q11 Provide copy of Traveler; this is not in BWR/6's TSIP conversion.

Status: Open

Discussion: *The requested traveller is provided.*

3.6Q12 Also deleted (c) should be revised for the contents of existing TS 4.4.2.4.b.

Status: Open

Discussion: *None of the containment isolation barriers specified in TS 4.4.2.4.b utilize a "pressurized sealing mechanism" as described in deleted item (c). Instead, these barriers use passive O-rings or other gasket designs which are normally maintained at atmospheric pressure. These barriers are designed such that if containment atmospheric pressure were to increase, then the sealing pressure against the O-rings is also increased to provide greater leak-tightness. Since item (a) already requires these barriers to be OPERABLE, item (c) is not required.*

3.6Q13 Background Item a.2 - The LCO name change is accepted but the preceding text change is rejected as not appropriate here in this LCO.

Status: Rejected

Discussion: *The text which is being deleted and replaced with "OPERABLE containment isolation barriers" is being removed for the following reasons. First, this text does not specify closed systems, end caps, and other types of containment barriers. Second, the bases state that all penetrations which are required to closed during accident conditions must be capable of being isolated by the containment isolation system or "closed by manual valves, blind flanges or de-activated automatic valves secured in their closed positions" (text in "" to be deleted). There are several penetration at Ginna System which are not isolated on a containment isolation signal, maintained open during normal operation, and which use check valves in combination with a closed system as the isolation barrier. This includes the main feedwater lines and charging lines to the RCS and RCPs. The proposed text clarifies that containment integrity is still met as long as the check valve and closed system for these penetrations is OPERABLE.*

58. ITS 3.6.2

i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers associated with subatmospheric, ice condenser and dual containment structures were deleted. This is an ITS Category (iv) change.
ACCEPTABLE

- ii. Note 2 for Conditions A and B was revised to provide additional clarification and consistency with the Condition statement. This is an ITS Category (iii) change.

[ITS58ii]:

3.6Q14

These text additions are redundant. Cannot be applied without Required Action notes already having this equipment inoperable as these text descriptions state has occurred. The place to add additional clarification is in the BASES.

Status: [] Rejected

Discussion: *RG&E agrees to withdraw this change to Required Action Note 2. However, the clarification will be maintained in the bases as suggested. Comment #36 has been opened to address this.*

- iii. The Note in the Frequency column for SR 3.6.1.2 was moved to the Surveillance column as preferred by licensed personnel. This is an ITS Category (iv) change.

[ITS58iii]:

3.6Q15

This "Note" change is rejected as not conforming to ITS format. This may be an ITS generic item in all chapters worthy of a generic solution.

Status: [] Rejected

Discussion: *Based on a telephone conversation with Carl Schulten on 8/23/95, the NRC has agreed that the relocation of Surveillance Frequency notes is a "good practice" and that either the NRC or the industry will generate a Traveller to incorporate. Comment #34 has been opened to track this resolution. Suggest change status to "open" for the interim.*

- iv. The Frequency for SR 3.6.2.2 was revised from 184 days following entry into containment to once every 24 months (i.e., once every refueling outage). The current Ginna Station TS do not contain a Surveillance for the air lock door interlock mechanism; however, RG&E believes that it is prudent to add a SR to ensure compliance with the specification. A Frequency of once every 24 months is considered appropriate since the interlock is purely mechanical and procedures are in place to control personnel access to containment during MODES 1 through 4. Also, this surveillance could challenge containment integrity if the interlock were to fail and both air lock doors were opened simultaneously. Finally, if the interlock is defeated during any shutdown condition, it must be retested prior to declaring it OPERABLE. Due to these changes, approved Traveller BWR-15, C.2 was only incorporated in part. This is a ITS Category (i) change.

[ITS58iv]: Also see [CTS#31.v-L1]

3.6Q16

In the improved TS SR 3.6.2.1, Appendix J requires Type B tests of the air lock and the door seals every 184 days. There are no specific existing TS requirements for testing the air lock interlock mechanism; however, TS 4.4.2.4.c implies opening of the air lock door to do seal tests every six months. Also, the Frequency of 184 days for SR 3.6.2.2 was selected as consistent with Appendix J testing intervals for air locks and for CIVs with resilient seals.

Status: [] Open

Discussion: Ginna Station personnel normally enter containment on a monthly basis to perform TS required surveillances. As such, the 184 day frequency in NUREG-1431 does not correspond to the use of the air locks themselves or Appendix J testing of the air locks (since the airlocks must be leak tested following entry). Also, see response to 3.6Q17 below.

3.6Q17 The explanation that testing the interlocks jeopardizes containment integrity is not accepted; otherwise the Condition B, Note #2 Relaxation of stationing a dedicated individual to maintain one door closed must be withdrawn. Verifying the interlock mechanism only prior to containment entry is needed to limit this SR; so not deleting the note helps reduce the frequency of this test. Without this exception, the air lock doors would be required to be opened solely to perform this interlock test. This scenario would then also require the door seal test be performed within the next 48 hours creating unnecessary containment entries and requiring manpower for testing. In the event the plant is utilizing one air lock for entries and maintaining one air lock idle, this surveillance would impose an excessive testing requirement.

Status: Open

Discussion: The reason that containment integrity could be challenged during the test of the interlock mechanism is that the only method of performing this test at Ginna Station is open one door and then attempt to open the second door. If the second door did open, you would then have a direct path to the outside environment until at least one door was closed. Note #2 to Condition B allows the use of an individual who in effect acts as the interlock mechanism if this mechanism is inoperable (i.e., this individual prevents opening more than one door at a time). As such, there is no direct relationship between the justification and Condition B Note. Relaxing the surveillance frequency of SR 3.6.2.2 to 24 months (i.e., each refueling outage) negates the need for the Note stating that the SR is only required prior to containment entry. If the frequency were maintained at 184 days, then this Note would be required. Please note that there is a WOG traveller in the system to make the change proposed in 58.iv.

3.6Q18 Also RG&E has 48 hours to test per the CTS and now has 72 hours per ITS. This is a relaxation which not been justified.

Status: Open

Discussion: The 72 hours to test an air lock door which has been opened is based on 10 CFR 50, Appendix J requirements. Hence, the CTS 4.4.2.4.c requirement of performing this test within 48 hours is more restrictive than Appendix J. Since Appendix J has been applied to all reactors, the allowance of 72 hours versus 48 hours has been shown to be generically acceptable. The reason that the CTS have a more restrictive requirement is related to the original Appendix J rule. For this rule, a leakage test was required after each opening. RG&E requested, and was granted, an exemption by the NRC to revise this requirement to only apply within 48 hours after the first in a series of openings (see letter from D.L. Ziemann, NRC, to L.D. White, RG&E, dated March 28, 1978). Therefore, RG&E proposes to eliminate this 48 hour testing requirement and follow the current requirements of Appendix J. [This response was revised per comment

#35]

- v. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Plant-specific design considerations were added with respect to the containment air locks.
 - b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.

[ITS58.V]:

3.6Q19

The deletion of the last sentence of the background needs explanation from traveler.

Status: [] Open

Discussion: *The requested traveller is provided.*

3.6Q20

P_o is 60 psig in existing TS, 59.8 psig in 3.6.1 and 59.3 psig here?

Status: [] Open

Discussion: *The design basis pressure of containment is 60 psig (see UFSAR 6.2.6.1). The resulting containment pressure post-LOCA is 59.3 psig while the containment pressure following a main steam line break is 59.8 psig. Therefore, using the definition of P_o from Appendix J of "the calculated peak internal containment pressure related to the design basis accident," the real value of P_o is 59.8. However, all leakage testing at Ginna Station has conservatively used the design pressure as P_o since the first CTS define P_o as "containment vessel design pressure" (see letter from D.L. Ziemann, NRC, to L.D. White, RG&E, dated March 28, 1978). The P_o value in the Applicable Safety Analysis bases is 59.8 psig though it could be read as 59.3 psig due to the copy quality (see Attachment C to the submittal). RG&E believes no change is required. [This response was revised per comment #35]*

3.6Q21

In LCO, recommend not adding text after OPERABLE or changing to "such that both doors are closed with leakage within acceptable limits."

Status: [] Open

Discussion: *The replacement text being proposed implies that if an airlock door is open, then the air lock must be declared inoperable. The text which was added essentially states that the air lock is OPERABLE if both doors can be closed with leakage within acceptable limits. The reason this text was added was to clarify that both doors did not have to be closed for the air lock to be OPERABLE. RG&E suggests leaving text as is or replace with the following: such that both doors are closed (except as being used for normal entry and exit from containment) with leakage within acceptable limits.*

3.6Q22

Explain why air locks not applicable in MODE 6.

Status: [] Open

Discussion: *RG&E has proposed to relocate the MODE 6 containment requirements to the TRM as discussed in Attachment A, Section C, item 107.i. [This*

response was revised per comment #221]

3.6Q23 Second insert to first paragraph of A.1,A.2 and A.3 rejected because indication lights only check not accepted for a physical "close and lock" action.

Status:[] Rejected

Discussion: The first bases paragraph of Required Actions A.1, A.2, and A.3 discusses the action to close the airlock only. The actual locking of the air lock is discussed in the following paragraph. Therefore, the insert in question states that the verifying the control board alarms for the air lock doors is adequate to verifying that the door is closed within 1 hour. The door must then be physically locked within 24 hours. This is no different than closing an inoperable containment isolation barrier from the control room and verifying from indicating lights that it is closed within 4 hours.

3.6Q24 The word clarification to Note #1 of Condition A and B does not clarify.

Status:[] Rejected

Discussion: The changes to Note #1 in the bases for Conditions A and B was made at the request of Ginna Station operations. The change in question revises the Note so that it reiterates the actual note in the Required Actions. The remaining text then clarifies the Note as required. This technique of reiterating the LCO note in the bases and then explaining as necessary is used throughout the ITS. There are only two clarifications made to Note #1 in the bases. The first is to add reference to Required Action C.3 for completeness since C.1 and C.2 are mentioned. The second is to state that even though Condition C is entered with both doors of an airlock inoperable, the tracking of the Completion Time in Condition A must still be performed in the event that one door is restored to OPERABLE status. This prevents misuse of the LCO whereby the time clock for Condition A would not start until after one door was restored providing an extra hour to isolate the remaining failed door.

3.6Q25 The new text insert from traveler for 58.viii needs explanation.

Status:[] Open

Discussion: The requested traveller is being provided.

3.6Q26 Third paragraph of B.1, B.2, and B.3 allow for procedure only administrative control is rejected. This means the area is locked and personnel are not permitted to enter.

Status:[] Rejected

Discussion: The "procedure control" text was added with respect to administrative controls since this is the definition of administrative controls at Ginna Station. That is, by use of procedural controls which restrict activities with respect to the inoperable airlock, it is ensured that personnel do not use it. The physical layout of Ginna Station does not allow for locking of an "area" related to an airlock (though the airlock doors can be physically locked).

3.6Q27 In C.1, C.2, and C.3, retain the "(eg., only one seal per door has failed)"

Status:[] Open

Discussion: RG&E agrees to retain the text in question. Comment #37 has been opened to address this.

3.6Q28 The references #3 can not be checked this should be referred to NRC PM to verify.

Status: [] Open

Discussion: To be discussed at the meeting.

3.6Q29 The deletion of the last sentence of SR 3.6.2.1 is not justified.

Status: [] Open

Discussion: RG&E agrees to retain the text in question. Comment #38 has been opened to address this.

3.6Q30 Issues raised in LCO review are also open but not identified here.

Status: [] Open

Discussion: To be discussed at the meeting.

- vi. Incorporation of approved Traveller BWR-16, C.20. Not checked
- vii. Incorporation of approved Traveller WOG-23, C.4. Not checked
- viii. Incorporation of approved Traveller BWR-16, C.24. Not checked
- ix. Note 3 was revised to provide consistency with LCO 3.6.3, Note 4. This is an ITS Category (iii) change. ACCEPTABLE, there should be a WSTS traveler prepared for this change!
- x. Required Actions C.2 and C.3 were revised to make air lock plural since more than one airlock may be affected when in Condition C. This change is also consistent with the bases for this Condition. This is an ITS Category (iii) change.

[ITS58.x]:

3.6Q31 This condition is governed by separate condition entry. So each airlock is on a separate clock. With the change, it could be misinterpreted that if both air locks were not restored to operable status at the same time then a plant shutdown is required. If the plural word in the BASES is misleading at your plant, then you should propose a clarification in the BASES.

Status: [] Rejected

Discussion: RG&E agrees to withdraw change 58.x. Comment #39 has been opened to address this.

- xi. The airlock acceptance criteria was also revised to be $\leq 0.05 L_a$ for each airlock and $\leq 0.01 L_a$ for each door. These changes are consistent with current Ginna Station testing practices since airlock acceptance criteria are not specified in TS. This is an ITS Category (i) change.

[ITS58.xi]: Also see [CTS 31.v-L1]

3.6Q32 Clarify the basis for revising the above leakage rate criteria when there are no existing TS requirements. How is this determined to be acceptable? Additionally, the above does not discuss the other SR 3.6.2.2.a & .b test changes which are not justified. These changes

are identical to what was already per the NUREG-1431.

Status: [] Rejected

Discussion: The NUREG-1431 SR 3.6.2.2 has acceptance criteria for air locks as follows:

- a. Overall air lock leakage rate is $\leq [0.05L_p]$ when tested at $\geq P_o$.
- b. For each door, leakage rate is $\leq [0.01L_p]$ when tested at $\geq [psig]$.

The change proposed by 3.6Q32 is to clarify that item a. is the leakage limit for each airlock. That is, the Ginna Station PORC questioned whether item a. was the leakage rate for both air locks or was it the leakage rate for each airlock. Ginna Station currently applies this leakage rate for each air lock in NUREG-1431. Hence, the only "change" in question is if the interpretation is made that the leakage limit in a. applies to both air locks. The changes to item b. are to just reword the text so that it reads similar to the revised item a. (i.e., begins with "Leakage rate...") and to replace the text in []s. Since the text in []s means a plant specific value is to be used, replacing "psig" with " P_o " in []s is considered acceptable since it is similar to the acceptance criteria of item a.

59. ITS 3.6.3

- i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers associated with subatmospheric, ice condenser and dual containment structures were deleted. This is an ITS Category (iv) change. ACCEPTABLE
- ii. The title, LCO, Conditions, Surveillances, and bases were revised to replace "valves" with "barriers." This change is consistent with current Ginna Station TS 3.6.3 and the ITS bases which require valves, blind flanges, and closed systems to be OPERABLE. Since valves are only a subset of the devices which provide containment integrity, "barriers" is considered a more appropriate term (see Ref. 23). This is an ITS Category (iii) change.

[ITS59.ii]: Also see [CTS16.iii-L1]

3.6Q33

This name change is not consistent with the existing TS 3.6.3 which was containment isolation boundaries. The existing TS also included air locks under this definition. Air Locks have a separate LCO 3.6.2. The use of barrier would include air locks and this is not appropriate. "Barrier" would also apply to the containment structure under LCO 3.6.1 and this is not appropriate. LCO 3.6.3 is meant to apply to only containment isolation valves and only those devices which block the penetration flowpaths. It is noted that the licensee is not proposing to call the mini-purge valves the "mini-purge barriers". Reference #23 has not been made available for review.

Status: [] Rejected, 7-14-95

[] Open, 7-27-95

Discussion: This item was originally rejected but it was reopened following a telecon on 7-27-95. The name change from containment isolation "boundaries" to "barriers" is accepted for the title of LCO 3.6.3;

however, internally within the LCO each name change must be discussed on the following guidelines:

1. Accept LCO title changed in all locations.
2. Accept LCO statement changed.
3. Actions note #3, only the word "valves" changed accepted to "barriers".
4. Actions Note #4, change accepted.
5. Proposed word changes in all Conditions not accepted; however, new Conditions can be written for those "barriers" which do not meet current GDC requirements and have supporting documentation for NRC acceptance. Example separate those barrier requirements now located in Condition A into their own separate Condition. Don't delete the Condition Notes but revise and make extensive use of notes to each Condition to describe the differences.
6. Changes to old SR 3.6.3.3 and 3.6.3.4 not accepted; however, new SRs may be written to accompany new Conditions to make appropriate for certain types of barriers.
This is to be discussed at the meeting.

- iii. Note 3 was revised to provide clarity and consistency with the bases. This is an ITS Category (iii) change.

[ITS59.iii]:

3.6Q34 The text clarification added to Note #3 are rejected because the LCO does not have to be redundant to the BASES text. The BASES are the location where clarifications such as these are placed. The justification for ITS changes to be consistent with the BASES is backwards, wrong and should be rejected. See item #3 of ITS59.ii.

Status:[] Rejected

Discussion: To be discussed at the meeting.

3.6Q35 The Note #1 needs more wording. It is recommended that the bracketed phrase be either [except for the Shutdown Purge System valve flow paths...] or [except for the 36-inch shutdown purge system valve flow paths...]

Status:[] Open

Discussion: RG&E agrees to revise the text for Note #1 to use "except for the Shutdown Purge System valve flow paths..." Comment #40 has been opened to address this.

- iv. Conditions A and B were revised to become more generic and Condition C was not added. The ITS bases state that isolation devices are either active or passive and that closed systems provide a passive barrier. The bases also state that closed systems are required to be intact for normally closed containment isolation valves to be considered OPERABLE. However, the Conditions are organized based on penetrations which have containment isolation valves and penetrations which have closed systems. To ensure consistency with the bases, Conditions A and B were changed to apply to all penetrations. A new Required Action A.2 was also added which allows a closed system to be used to isolate a failed isolation barrier. This change now allows any device which must be OPERABLE to meet the LCO to be used to isolate a failed containment isolation barrier. This change addresses the issues discussed in Reference 24. A new Required Action B.2, similar to A.1.2, was also added as a result of

the above change. These are ITS Category (i) changes.

[ITS59.iv]:

3.6Q36

The NUREG-1431 LCO Condition A and B were to cover all 10 CFR50, Appendix A, GDC 55 and GDC 56 type of penetrations. Condition C was to cover GDC 57 type of penetrations. It is acknowledged that Ginna was designed prior to the 10CFR 50, Appendix A, GDC; however, the Ginna design is not clearly presented in the LCOs. Does Ginna have lists of the containment penetrations to identify all those penetrations which meet the GDC and those which don't. Those penetration which meet GDC should go into Conditions A, B and C. Propose new Ginna Conditions for those which don't meet GDC requirements.

Status:[] Open

Discussion: *To be discussed at the meeting.*

3.6Q37

Condition A and B were developed to tie together. Insert 3.6.3.2 is not needed because when two valves are inoperable in Condition B and one is restored operable you are back into Condition A, Required Action A.2 which is the same as the proposed B.3 (except for the new Completion Time "devices outside of containment" text that is not acceptable). Proposed Required Action B.2 is rejected for the same reason that insert 3.6.3.10 to BASES is acceptable. Per the GDC, you cannot isolate a penetration open to containment with just one (unreliable) device (eg. check valve or a closed system) which is a leaking barrier needing periodic leak rate evaluation. Condition C was only written for one inoperable containment isolation valve on a "closed system inside of containment" do not write this into Condition A. Has the NRC staff accepted alternate positions for the Ginna design which are different from the GDC? If so, keep Condition C separate and rewrite these alternates in new Conditions for the Ginna design.

Status:[] Rejected

Discussion: *To be discussed at the meeting.*

3.6Q38

Proposed A.2 (insert 3.6.3.1) and its associated changes are rejected because closed system operability is determined by performance of Type A, B, and C tests and not by reliance on water leakage detection systems, operator walkdowns and other surveillance systems as described in the BASES. It is very awkward (if not impossible) to rewrite Condition A and B to do the same as a separate Condition C. If Condition C is used, it permits only the operability of the same affected closed system penetration to act as a second barrier once the sole inoperable isolation valve in the affected penetration is isolated.

Status:[] Rejected

Discussion: *To be discussed at the meeting.*

3.6Q39

Condition A and B disagree. Condition A exempts the mini-purge valve and Condition B exempts all purge valves? What goes on here? Also what condition is entered if the shutdown purge valve inoperable? How is it different from the mini-purge valve? Providing piping schematics with the containment boundary indicated would help this evaluation. The notes to Conditions A and B could be deleted if they were written as developed but the note to

Condition C should stay where ever Condition C ends up being located. Add more Conditions, if needed.

Status:[] Open

Discussion: *To be discussed at the meeting.*

- v. Incorporation of approved Traveller BWR-15, C.15. ACCEPTABLE
- vi. Condition D, SR 3.6.3.11, and the associated bases were not added since Ginna Station does not have a shield building. This is an ITS Category (iv) change. As such, approved Traveller BWR-14, C.3 and C.4 were not incorporated. ACCEPTABLE, but contents of traveler was not reviewed.
- vii. SR 3.6.3.1 and the associated bases were not added since the Shutdown Purge System is isolated by a blind flange (see Ref. 25). The LCO bases were revised to reflect that the blind flange must be installed for the containment isolation barrier for the Shutdown Purge System to be considered OPERABLE. Verification that this blind flange is in place is accomplished by new SR 3.6.3.2. This is an ITS Category (i) change. As such approved Traveller NRC-02, C.21 was not incorporated.

[ITS59.vii]:

3.6Q40

The existing TS 4.4.2.4.b requirements to assure the shutdown purge valves are operable needs to be included in the improved TS. The 36-inch shutdown purge valves are used for ventilation of containment below MODE 4 and prior to personnel access. The valves must be operable at these times, during refueling and the flowpaths unrestricted by blind flanges. Where are the surveillance requirements (if not SR 3.6.3.1) and/or leakage rate tests for these valves (if not SR 3.6.3.7) to determine operability? Furthermore, what Conditions are entered until these valves are restored to service?

Status:[] Open

Discussion: *LCO 3.6.1 covers containment isolation requirements in MODES 1, 2, 3, and 4. The shutdown purge valve flowpaths are isolated with flanges containing double O-rings during these MODES and are only opened when below MODE 4. As such, the flanges provide both required containment barriers above MODE 5 and the valves are not required, nor subject to, Appendix J leakage testing (see Background bases for Shutdown Purge System). The LCO bases state that "both penetrations associated with the Shutdown Purge System must be isolated by a blind flange containing redundant gaskets, or a single gasketed blind flange with a de-activated isolation valve (i.e., two passive barriers)." Therefore, if the flanges are not in place above MODE 5, two barriers for the affected penetration are declared inoperable. The flanges are considered Type B isolation barriers and tested in accordance with Appendix J. The only accident in which the shutdown purge valves are required below MODE 4 is during refueling operations. This is addressed in Attachment A, Section C, item 107.i.*

- viii. SR 3.6.3.2 and the associated bases were not added since this surveillance is not in the current Ginna Station TS. The Background bases have been revised to state that "emphasis shall be placed on

limiting purging and venting times to as low as reasonably achievable." All uses of the Mini-Purge System are under procedural control. In addition, the status of the mini-purge isolation valves is continuously available in the control room such that operators would be quickly aware of any valve that is not closed. Verification of these status lights is performed daily by operators such that a Surveillance every 31 days is unnecessary. This is an ITS Category (i) change. As such, approved Traveller BWR-15, C-19, Revision 1 was not incorporated.

[ITS59.viii]:

3.6Q41

The SR 3.6.3.2 seems to be identical to existing TS 3.6.5, Containment Mini-Purge. This existing TS requirement is now a surveillance instead of an LCO. The SR and BASES should not be deleted as proposed. Also, please note that this SR is a visual verification and not just satisfied by checking the status of the indication lights.

Status: [] Rejected

Discussion: CTS 3.6.5 states as follows:

Whenever the containment integrity is required, emphasis will be placed on limiting all purging and venting times to as low as achievable. The mini-purge isolation valves will remain closed to the maximum extent practicable but may be open for pressure control, for ALARA, for respirable air quality considerations for personnel entry, for surveillance tests that may require the valve to be open or other safety related reasons.

There are no surveillance requirements every 31 days with respect to verifying that the mini-purge valves are closed as required by NUREG-1431 SR 3.6.3.2. The conditions for which the mini-purge valves can be opened do not meet any of the four criteria since the valves receive containment isolation signals and are designed to close within 2 seconds. These restrictions are instead good practices. In addition, the proposed wording of ITS 3.6.3.1 and ITS 3.6.3.2 provide for necessary verification of mini-purge valve position without having a separate SR. Therefore, RG&E does not believe that SR 3.6.3.2 is required. [This response was changed during meetings week of 10/9/95. See comment #126.]

3.6Q42 Is the traveler applicable now?

Status: [] Open

Discussion: *The requested traveller is being provided. This was approved on 6/2/94.*

- ix. SR 3.6.3.3 and SR 3.6.3.4 have been revised to clarify that this verification is performed to ensure that nonautomatic isolation barriers which are required to be closed immediately following an accident are in fact closed, versus ensuring isolation barriers "closed during accident conditions are closed." Since several penetrations are normally open but are isolated during accident conditions if the first passive barrier fails, the existing SR wording is misleading. Also, the SR Frequency was revised from 31 days to 184 days consistent with Ginna Station practices. In addition, this SR was revised to not require verification of isolation barriers which are locked, sealed closed, or otherwise isolated similar to other Surveillances. The current Ginna Station

TS do not contain this requirement. However, all containment isolation barriers have a special field tag identifying the device as an isolation barrier and specifies that Operations must be notified before changing the position of the device. This tag provides sufficient administrative controls such that a Frequency of 184 days is considered adequate. These are ITS Category (i) changes.

[ITS59.ix]:

3.6Q43 This word clarification does not help, as noted in item #11 of ITS 59.ii.

Status: [] Open

Discussion: *To be discussed at the meeting.*

3.6Q44 Since the existing TS has no equivalent to SR 3.6.3.3 and 3.6.3.4, the Frequency of 31 or 92 days maintains a consistency with the periodic check required when in the various Conditions of this LCO depending whether the valves and blind flanges are inside or outside containment. These intervals were not changed from the previous STS and are consistent for all five Owner Group improved STS developed. It was determined during the development of the improved STS that this SR did not impose an unnecessary hardship. It was believed that the more consistent requirements were used, the less chance existed for missing an important SR. The tagging system is good but strictly independent of the Frequency for this SR.

Status: [] Open

Discussion: *The Required Actions within LCO 3.6.3 for inoperable containment barriers require isolation of the penetration within 1-4 hours and verification every 31 days that the penetration remains isolated. This verification every 31 days is considered a "penalty" since the containment barrier originally credited in the accident analysis is inoperable such that alternate measures had to be taken. This is similar to other LCOs which require increase surveillances with inoperable equipment. Therefore, requiring verification every 31 days that OPERABLE containment isolation barriers are in their correct position is considered excessive. These barriers are strictly controlled at Ginna Station with clearly identified tags located on them such that they require Shift Supervisor clearance to operate and subsequent independent verification. The only time in which these valves are opened is during IST related testing. Since this testing will now be on a quarterly basis, RG&E is willing to agree to quarterly verification of all valves outside containment to match up with the testing. For those valves inside containment, RG&E will agree to the NUREG-1431 frequency during startup if not within the last 92 days. [This response was changed as a result of meetings the week of 10/9/95. See comment #123.]*

- x. SR 3.6.3.5 and the bases were revised to remove verification of "each power operated" containment isolation valve's isolation time. This SR is performed to ensure that those containment isolation valves which receive a containment isolation signal to automatically close are tested in accordance with the IST program. At Ginna Station, several power operated containment isolation valves do not receive a containment isolation signal. Therefore, the isolation time of these valves is not relevant to this LCO. The change also

provides consistency with new SR 3.6.3.5. This is an ITS Category (i) change. ACCEPTABLE

- xi. SR 3.6.3.6, SR 3.6.3.9, and the associated bases were not added since these tests are only required for plants with subatmospheric containments which does not apply to Ginna Station. This is an ITS Category (iv) change. ACCEPTABLE
- xii. SR 3.6.3.7 and the bases were revised to provide consistency with SR 3.6.2.1. The SR text was also changed since it only applies to the Mini-Purge System as the Shutdown Purge System is isolated above MODE 5 per the new LCO bases. The specified Frequency was revised since the requirement for more frequent testing of the mini-purge isolation valves was removed from the Ginna Station TS by Amendment No. 54 (Ref. 23). This is an ITS Category (i) change. Approved Traveller BWR-14, C.3 was also incorporated.

[ITS59.xii]:

3.6Q45 The SR requirements for Shutdown Purge System valves at refueling should be located here or in ITS 3.9.4; but it is deleted. Explain?

Status: [] Open

Discussion: *RG&E has proposed to relocate the MODE 6 containment requirements to the TRM as discussed in Attachment A, Section C, item 107.i. [This response was revised as a result of 11/16/95 Appeal Meeting. See comment #221.]*

3.6Q46 What is in Amendment 54 that is applicable here? Existing TS 4.4.2.4.a is very general and does not apply here.

Status: [] Open

Discussion: *CTS 4.4.2.4.a used to have a requirement to test the mini-purge valves every 184 days. However, this requirement was removed by Amendment 54 (see letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, dated August 30, 1993). Therefore, the more frequent surveillance of the mini-purge valves does not apply to Ginna Station.*

3.6Q47 The Frequency Column should have an interval of 184 days and within 92 days after opening the valve.

Status: [] Open

Discussion: *See response to 3.6Q46 above. No change is required.*

xiii. Incorporation of approved Traveller NRC-03, C.9, Revision 1. ACCEPTABLE

xiv. SR 3.6.3.10 and the associated bases were not added since the Shutdown Purge System is isolated in MODES 1, 2, 3, and 4 by a blind flange. The NUREG-1431 bases state that this SR only applies to plants which can have the shutdown purge valves open above MODE 5. Therefore, this SR is not applicable to Ginna Station. This is an ITS Category (iv) change. As such, approved Traveller WOG-11, C.1 was not incorporated. ACCEPTABLE

xv. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including providing consistency with current Ginna Station TS bases.
- b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.

[ITS59.xv]:

3.6Q48

There are 28 pages of bases with changes. Each page has changes amounting to over one hundred changes. If these Conditions are rewritten, as requested, then the BASES will have to be significantly changed. The BASES will have to be discussed later separately rather than now. Also, providing an itemized list of comments now will result in most not being applicable.

Status: [] On Hold

Discussion: *To be discussed at the meeting.*

- xvi. Incorporation of approved Traveller BWR-15, C.9. Not checked
- xvii. Incorporation of approved Traveller BWR-16, C.22. Not checked
- xviii. Incorporation of approved Traveller BWR-15, C.5. Not checked
- xix. Incorporation of approved Traveller WOG-11, C.2. Not checked
- xx. The LCO was revised to add a Note stating that the main steam isolation valves, main steam safety valves, and atmospheric relief valves are not included in this LCO when they are required to be OPERABLE in Chapter 3.7. The valves all credit the SG tubes as a boundary such that additional time is allowed to restore OPERABILITY. This change is consistent with the bases for ITS Chapter 3.7. This is an ITS Category (i) change.

[ITS59.xx]:

3.6Q49

At Ginna, where is the containment integrity boundary with respect to these valves? Please provide a sketch.

Status: [] Open

Discussion: *The steam generators act as the first boundary for these penetrations with the valves providing the second boundary. The UFSAR drawings for these affected penetrations are attached (UFSAR Figures 6.2-76, -77, and -78).*

3.6Q50 Are these valves Type C tested?

Status: [] Open

Discussion: *No, these valves are not Type C tested. These valves are in the main steam and feedwater lines which only require Appendix J testing in BWRs. In addition, the main steam safety valves and atmospheric relief valves are relief valves whose setpoint is greater than 1.5 times the design pressure of containment. Leak testing of these valves was addressed during the amendment request to relocate the listing of CIVs from technical specifications (see letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, dated November 30, 1992).*

3.6Q51 Why doesn't Actions Note #3 apply in these cases?

Status: [] Open

Discussion: Actions Note #3 requires entry into any applicable Conditions and Required Actions by the inoperable containment isolation barrier. The Required Actions related to the main steam isolation valves main steam safety valves and atmospheric relief valves are less restrictive than LCO 3.6.3. If 8 hours or even 7 days has been justified for restoring these valves to OPERABLE status in their respective LCOs, why should LCO 3.6.3 be more restrictive. This is especially true since the only accident which can challenge the first barrier for these penetrations is a steam generator tube rupture (SGTR) at which time containment is bypassed. The dose analyses have been performed with a stuck open safety relief valve in this instance and demonstrated acceptable results.

3.6Q52 Alternately, why can't these exemptions be explained in the BASES, so this note could be eliminated from the LCO statement?

Status: Open

Discussion: These valves are identified as containment isolation barriers in the UFSAR and station procedures. Therefore, if these valves are inoperable from a containment isolation standpoint, LCO 3.6.3 must be entered since we cannot remove their containment isolation function under 50.59. Placing this type of information in the bases would leave the window open for future questions which is best addressed now. There is also a WOG traveller in the system for this change.

xxi. Condition E was revised to provide consistency with LCO 3.6.2. The mini-purge valves at Ginna Station have similar leakage acceptance criteria to the containment air lock doors. As such, failure of one mini-purge valve does not require evaluation with respect to overall containment leakage. However, failure of both valves does require consideration of containment leakage since the penetration no longer meets its leakage criteria as specified in new SR 3.6.3.4. Therefore, Condition E was revised to apply to one mini-purge valve not within leakage limits and a new Condition was added for two valves not within leakage limits. In both of these Conditions, Required Action E.3 was not added since Ginna Station currently does not have this requirement. Also, due to the design of the mini-purge penetrations, it may not be possible to test a mini-purge valve if the second in-series valve is excessively leaking. These are ITS Category (iii) and (i) changes.

[ITS59.xxi]:

3.6Q53. Why compare this to LCO 3.6.2?

Status: Open

Discussion: Required Action C.1 of NUREG-1431 LCO 3.6.2 requires assessment of containment leakage per LCO 3.6.1 if both doors of an air lock are inoperable. The proposed Condition D of ITS 3.6.3 requires this same assessment if both mini-purge valves for a penetration are not within their leakage limits.

3.6Q54 Where is the existing TS acceptance criteria for air locks?

Status: Open

Discussion: As discussed in the response to 3.6Q32, there are not current TS acceptance criteria for the air locks.

3.6Q55 Do the mini-purge valves have resilient seals or not?
Status: [] Open
Discussion: *The mini-purge valves do use resilient seals. However, as noted in the response to 3.6Q46, more frequent testing of these valves is not required.*

3.6Q56 The noted inability to these valves to hold a test pressure gives concern for their capability to isolate as a deactivated automatic valve.

Status: [] Open
Discussion: *The 59.xxi discussion does not state that the mini-purge valves cannot hold test pressure. The only method to test the first valve located inside containment is to perform a reverse flow test (i.e., pressurize between the two mini-purge valves as shown on UFSAR Figures 6.2-41 and -66). Therefore, if the mini-purge valve located outside containment is leaking, you cannot test the first valve unless the leaking valve was first repaired. The mini-purge valves have continued to demonstrate excellent leak tightness which is the basis for removing the more frequent testing as discussed in the response to 3.6Q46.*

60. ITS 3.6.4

i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers (including LCO 3.6.4B) associated with subatmospheric, ice condenser and dual containment structures were deleted. This is an ITS Category (iv) change. ACCEPTABLE

ii. The Completion Time for Required Action A.1 was revised from 1 hour to 24 hours consistent with current Ginna Station TS 3.6.2. Allowing 24 hours to restore pressure to within limits is acceptable due to the large containment free volume and limited size of the containment Mini-Purge System. This is an ITS Category (i) change.

[ITS60.ii]: Also See [CTS16.ii-L1]

3.6Q57 Explain how containment pressure is affected by the limiting size and function of Mini-Purge System.

Status: [] Open

Discussion: *The time to depressurize containment is dependant upon the initial containment pressure and the size of the Mini-Purge System opening to the outside environment. Consequently, with a larger vent path from containment, the faster the depressurization if the Mini-Purge System is available.*

3.6Q58 How long does it take to open the isolation valves to return to atmospheric pressure?

Status: [] Open

Discussion: *The time to depressurize containment is dependant upon the initial containment pressure. Ginna Station normally begins to depressurize after containment reaches 0.5 psig since the upper limit is only 1 psig. Depressurization to atmospheric pressure under these conditions normally takes less than 30 minutes.*

3.6Q59 Is this pathway filtered?

Status: [] Open

Discussion: The pathway through the mini-purge exhaust exits into the Auxiliary Building charcoal filter units. However, these charcoal filters are not ESF components, and as such, are not credited in the accident analysis. Instead, the mini-purge valves are designed to close within 2 seconds following receipt of an isolation signal.

3.6Q60 How long does it take to exchange containment air volume?

Status: [] Open

Discussion: The mini-purge system is designed to supply and exchange containment air volume at a rate of 1200 cfm (see UFSAR 6.2.4.4.9). With a containment air volume of 1,000,000 ft³, it takes approximately 14 hours to achieve one containment air volume exchange.

3.6Q61 Why 24 hours and not a Completion Time of 1, 2, 4 or 8 hours? Remember the original basis for 1 hour is consistency with the loss of the new containment operability per ITS 3.6.1!

Status: [] Open

Discussion: The 24 hours that was proposed is based on CTS 3.6.2 requirements. However, RG&E is willing to reduce this Completion Time to 8 hours since the time to reduce containment pressure is minimal assuming the Mini-Purge System is available. This 8 hours is still greater than the 1 hour for LCO 3.6.1 and is based on the following considerations. The preferred method of reducing containment pressure during MODES 1, 2, 3, and 4 is via use of the mini-purge valves. If any of these valves failed to open, providing 1 hour to repair the valves is a very short period of time, especially on the back shift. The alternate methods require use of small sample lines, etc. RG&E also anticipates a future TS amendment to increase the initial allowed containment pressure such that longer depressurization times could result. Finally, the Ginna Station containment structure has been pressure tested to 1.15% of its design rating which provides additional margin in the unlikely event of a DBA during this 8 hour window. [Comment #127 was opened to use 8 hours]

iii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including providing consistency with current Ginna Station TS bases.
- b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.

[ITS 60.iii.a and b]:

3.6Q62 There are several places where reference to inadvertent containment spray has been removed without explanation? Please explain.

Status: [] Open

Discussion: To be discussed at the meeting.

3.6Q63 There are changes for travelers in 60.iv and 60.v which are not available. Please explain.

Status: [] Open

Discussion: The requested traveller is being provided.

3.6Q64 Page B 3.6-45 is printed off to the side of the page please provide a new page.

Status: [] Open

Discussion: The requested page is being provided.

3.6Q65 In two places, the lower pressure limit is based on the requirements for the reactor coolant pump motors. Could you elaborate?

Status: [] Open

Discussion: To be discussed at the meeting.

3.6Q66 The changes for the 24 hour Completion Time are still under review in the LCO.

Status: [] Open

Discussion: Please see response to 3.6Q61.

3.6Q67 Is the exact reference to PI-944 in SR 3.6.4.1 necessary? Shouldn't this instead be a BASES reference document? Lastly, doesn't new paragraph insert belong in Bases for LCO in Section 3.3 and not here?

Status: [] Open

Discussion: Since SR 3.6.4.1 is a new surveillance requirement for Ginna Station, reference to PI-944 is provided to ensure the correct pressure indication is used for this surveillance. This is the only containment pressure indicator which has the necessary tolerance and control room indicator range to measure pressures between -2.0 psig and 1.0 psig. This level of detail in the bases was specifically requested by the Ginna Station PORC. With respect to the additional text related to the calibration requirements of PI-944, this pressure indicator is not used for any RTS, ESFAS, or PASS function. Consequently, Section 3.3 does not apply for this instrument.

iv. Incorporation of approved Traveller WOG-11, C.3. Not checked

v. Incorporation of approved Traveller WOG-11, C.1. Not checked

61. ITS 3.6.5

i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers (including LCO 3.6.5B and LCO 3.6.5C) associated with subatmospheric, ice condenser and dual containment structures were deleted. This is an ITS Category (iv) change. ACCEPTABLE

ii. The Completion Time for Required Action A.1 was revised from 8 hours to 24 hours. The current Ginna Station TS do not have a requirement for average containment air temperature. Since the Frequency for verifying that the average temperature is $\leq 120^{\circ}\text{F}$ is 24 hours, RG&E believes that 24 hours to restore the temperature to within limits is appropriate. A Completion Time of 24 hours is also consistent with new LCO 3.6.4 (and current Ginna Station TS 3.6.2). This is an ITS Category (i) change.

[ITS61.ii]: See also [CTS16.iv-M1]

3.6Q68 Explain the referred method for returning the air temperature to within limits?

Status: [] Open

Discussion: The only method to restore containment air temperature to within limits is to operate additional containment recirculation fan cooler (CRFC) units or increase Service Water flow to the operating fan coolers. All four CRFC units are required to be OPERABLE per ITS LCO 3.6.6; however, one or two fan coolers can be removed from service for up to 7 days. During summer months, all CRFC units are normally required to maintain containment temperature within acceptable limits. Therefore, Ginna Station could be in ITS LCO 3.6.6 with one or more CRFC coolers inoperable during the summer months.

3.6Q69 Does any light, alarm etc. notify the operators that the limits are being exceeded?

Status: [] Open

Discussion: There is no control room alarm or annunciator that global containment air temperature limits are being exceeded. However, there are individual alarms with respect to individual fan coolers associated with other containment ventilation systems (e.g., shroud fan coolers). The plant computer contains temperature readings within containment that are verified by operators every shift.

3.6Q70 What is the normal operating range of temperature? If restorative action were taken sooner before reaching the limits, why couldn't the Completion Time be 8 hours?

Status: [] Open

Discussion: The normal operating range during the winter is 90°F and during the summer, 110° - 115°F. If a CRFC was out of service during the summer, the containment air temperature could potentially exceed the 120°F limit by several degrees. The required cooldown below this limit would take several hours and is dependent upon the ability to restore the inoperable CRFC and SW temperature and flow.

3.6Q71 What if the SR 3.6.5.1 was performed every 12 hours?

Status: [] Open

Discussion: SR 3.6.5.1 is currently performed every shift via check of the plant computer. RG&E is willing to reduce this SR frequency if the Required Action Completion Time was subsequently increased. [Comment #128 was opened to revise the SR frequency to 12 hours and leave the Completion Time at 24 hours]

iii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including providing consistency with current Ginna Station TS bases.
- b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.

[ITS 61.iii.a and b]:

3.6Q72 The insert into the first sentence of Applicable Safety Analyses is not understood.

Status: Open

Discussion: The NUREG-1431 Applicable Safety Analysis bases for LCO 3.6.5 states that the containment average air temperature "establishes the containment environmental qualification operating envelope for both pressure and temperature." However, environmental qualification (EQ), by itself, is not a component OPERABILITY issue per Generic Letter 91-18. It is a qualification issue which must be evaluated similar to seismic considerations. As such, the EQ basis would not meet any of the four criteria for inclusion within TS. The inserted text clarifies that the containment average air temperature limit ensures that the energy within containment following a DBA is within the capacity of the CS and CRFC units such that containment integrity is maintained. The text also states that containment average air temperature is an important contributor with respect to EQ.

3.6Q73 The sentence at the end of the second paragraph of Applicable Safety Analyses is meant to be filled in with a description of the worst case single active failure for Ginna. Provide the worst case example.

Status: Open

Discussion: The worst case single active failure for Ginna Station depends on the accident scenario and the issue being considered (e.g., dose consequences, PCT, DNBR). In general, the worst case single active failure is the loss of one electrical train following a loss of offsite power. However, for a steam line break, the worst case failure is the loss of a safety injection pump since offsite power is assumed available to maintain the RCPs operating and generate a more rapid cooldown (i.e., there is no single failure which can render inoperable one electrical train). Therefore, this text was removed from the bases rather than attempt to address every possible worst case single failure.

3.6Q74 The phrase "exceed the containment design temperature" has been removed in two places but the sentence structure is lost and has to be rewritten to complete the sentence. Also the ending sentence of this paragraph is not justified for the new wording. Please explain.

Status: Open

Discussion: The phrase "exceed the containment design temperature" was removed in Attachment D to the submittal but replacement words contained in Attachment C were mistakenly not added. Comment #41 has been opened to correct this. The last sentence of the affected paragraph is based on UFSAR Section 6.2.1.2 as referenced earlier in the Applicable Safety Analyses bases. This issue is also discussed in UFSAR Section 3.11.3.

3.6Q75 Please explain the deletion of the paragraph on inadvertent actuation of the CS System.

Status: Open

Discussion: To be discussed at the meeting.

3.6Q76 Please explain the reason for adding the last phrase to SR 3.6.5.1.

Status: []
Response:

Shouldn't this be in an LCO in Chapter 3.3? Why here?

Open

The six temperature indicators identified in the bases are not used for any RTS, ESFAS, or PASS function (although they are RG 1.97, Type D instruments). Consequently, Section 3.3 does not apply for these instruments and the bases text was added.

62. ITS 3.6.6

- i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers (including LCO 3.6.6B, LCO 3.6.6C, LCO 3.6.6D and LCO 3.6.6E) associated with subatmospheric, ice condenser and dual containment structures were deleted. Since the Containment Spray System at Ginna Station is credited for iodine removal, LCO 3.6.6A was chosen. This is an ITS Category (iv) change. ACCEPTABLE
- ii. The LCO title, Conditions, Surveillances, and bases were revised to replace "Cooling System" with "Recirculation Fan Cooling System" consistent with Ginna Station nomenclature. Also, the Post-Accident Charcoal System was added to this LCO for reasons described below. This is an ITS Category (iv) change. ACCEPTABLE
- iii. Conditions C and E were revised, Condition D was deleted, and three new Conditions were added with respect to inoperable containment recirculation fan cooling (CRFC) units and post-accident charcoal filter trains. These changes were necessary due to plant-specific design features relative to the CRFC units that differ from the model plant used to produce NUREG-1431. At Ginna Station, there are four CRFC units which are all supplied by a single Service Water (SW) loop header (i.e., the SW system is only organized into trains at the pump level, not at the loop level). In addition, two of the four CRFC units (i.e., units A and C) connect to the Post Accident Charcoal System which does not have its own separate fan system. Consequently, if either CRFC unit A or C is inoperable, then the associated post accident charcoal filter train is inoperable such that Condition D cannot apply to Ginna Station (i.e., one would have to enter LCO 3.0.3). In addition, any one of the following combinations is successful for radioactive iodine removal post accident:
 - a. Two containment spray (CS) trains;
 - b. One CS train and one post-accident charcoal filter train; or
 - c. Two post-accident charcoal filter trains.

However, since at least one CS train must be OPERABLE above MODE 5 for containment pressure and temperature control, the last combination is not used. As such, organizing this LCO by trains for the Containment Recirculation Fan Cooling System and separating out the function of the Spray Additive and Post-Accident Charcoal Systems is not possible. Therefore, a new Condition (i.e., B) was added which allows one post-accident charcoal filter train to be inoperable for up to 7 days since at least one redundant post-

accident charcoal filter and one CS train is available. A second new Condition (i.e., C) was added for the case with two post-accident charcoal filters inoperable which requires that they be restored to OPERABLE status within 72 hours consistent with Condition A for loss of one CS train. A third Condition (i.e., D) was added with respect to an inoperable spray additive tank since this renders the CS iodine removal capability inoperable. A Completion Time of 72 hours is provided for this Condition also. In addition, existing Condition C was revised to address the case of one or two inoperable CRFC units and Condition E was revised to reflect all the possible combinations which result in the need to enter LCO 3.0.3. A Note was added to the LCO to require declaring the associated post-accident charcoal filter train inoperable when CRFC unit A or C is inoperable. Also, the necessary Surveillances and bases associated with the Post-Accident Charcoal System and the spray additive tank were added. These are ITS Category (i) changes.

[ITS62.iii]:

3.6Q77 The LCO statement has the word containment which is redundant to CRFC and should be deleted.

Status: [] Open

Response: *The LCO statement in Attachment D should have "containment" removed prior to CRFC units as shown in Attachment C. Comment #41 has been opened to correct this.*

3.6Q78 The new note under applicability is a Condition statement and Required Action. It is acknowledged but not fully understood. Please provide a sketch of the components and their connections. This will enable new Condition F to be rewritten with this note in it (and Required Actions added), another Condition may be created or some other solution implemented.

Status: [] Open

Response: *A simplified sketch of the relationship between the CRFCs and the Postaccident Charcoal filters is being provided. RG&E proposes to relocate this Actions Note to ITS Condition F as follows. Comment #42 has been opened to address this.*

<u>Condition</u>	<u>Required Action</u>	<u>Completion Time</u>
F. One or two CRFC units inoperable.	F.1-----NOTE----- Required Action F.1 only required if CRFC unit A or C is inoperable.	
	----- Declare associated post-accident charcoal filter train inoperable.	Immediately

AND

F.2 Restore CRFC unit(s) to OPERABLE status. 7 days

3.6Q79 It is acceptable to delete Condition D.

Status: [] Closed

Response: N/A

3.6Q80 Condition C needs a qualifier statement...that both CS trains are OPERABLE. This is because you could be in Condition A and C at the same time and not be able to meet the design accident.

Status:[] Open

Response: *Condition H specifically addresses the case of being in both Condition A and C at the same time and requires entry into LCO 3.0.3. Condition C was not written to only apply if both CS trains were OPERABLE since if a CS train were inoperable, then LCO 3.0.3 is entered. If the CS train is then restored to OPERABLE status, the time clock for Required Action C.1 would not begin until after the repair is completed instead of from the time that the post-accident charcoal filters were declared inoperable as it is currently written.*

3.6Q81 Isn't new Condition D a subset of Condition A? Please comment especially regarding how it affects both CS trains not being fully OPERABLE (no iodine removal capability). This assumes there is only one spray additive tank. Also this needs to say that both post-accident charcoal filters must be OPERABLE per CTS 3.3.2.2.e. The CTS refers to the spray additive system and you only spec the tank?

Status:[] Open

Response: *A sketch of the interaction of the CS trains and the single spray additive tank is provided which shows the breakdown of components as described in the LCO bases. As can be seen from this drawing, Condition D is not a subset of Condition A since the inoperability of the spray additive tank does not fail the CS pressure reducing function (i.e., it only impacts the iodine removal function). However, if both CS spray trains are inoperable, then Condition H requires immediate entry into LCO 3.0.3 regardless of the status of the spray additive tank. Condition H is also entered if the spray additive tank and one or both post-accident charcoal filters are inoperable. Condition D was not written to only apply if both post-accident charcoal filters were OPERABLE since if a post-accident filter were inoperable, then LCO 3.0.3 is entered. If the post-accident filter is then restored to OPERABLE status, the time clock for Required Action D.1 would not begin until after the repair is completed instead of from the time that the spray additive tank was declared inoperable as it is currently written. Also, note Comment #2 which has been opened related to a LCO 3.0.6 exemption for the spray additive tank and CS trains interaction.*

3.6Q82 New Condition E statement does not include Condition D if the Completion Time is not met.

Status:[] Open

Response: *The Condition E statement in Attachment D should have include Condition D as shown in Attachment C. Comment #41 has been opened to correct this.*

3.6Q83 In new Condition H couldn't the last three OR statements be shortened to "Any combination of three or more inoperable CRFC units, spray additive tank, CS train or post-accident charcoal filter trains.

Status:[] Open

Response: *No, the OR statement must be as stated since there are several combinations of the proposed text which would not be outside the*

accident analysis including:

1. The spray additive tank and one CS train.
2. One CS train and one post-accident charcoal filter.

3.6Q84 Couldn't we get rid of the Required Actions and Completion Time of Condition H and move the Condition H statements to below the Condition G statement and thus go directly to the Condition G Required Actions?

Status: [] Open

Response: *The purpose of Condition G is to require a plant shutdown since even though the plant is within the accident analysis assumptions, the required redundancy for the CRFC units is unavailable and was not restored within specified times. Condition H is provided in the case where the plant is outside the accident analysis assumptions since a loss of safety function has occurred and LCO 3.0.3 must be entered immediately. Also, entering LCO 3.0.3 provides the plant with 1 hour to prepare for the shutdown which Condition G does not.*

3.6Q85 SR 3.6.6.1 needs to also verify the spray additive valves.

Status: [] Open

Response: *RG&E agrees to add a new SR similar to NUREG-1431 SR 3.6.7.1. Comment #43 has been opened to address this.*

3.6Q86 New SR 3.6.6.13 should include the phrase "that is not locked, sealed, or otherwise secured in position" as justified in ITS #62.v.

Status: [] Open

Response: *RG&E agrees to add this wording to SR 3.6.6.13. Comment #44 has been opened to address this.*

3.6Q87 New SRs 3.6.6.5 and 3.6.6.6 may need to be renumbered per the Writers Guide. What is the real frequency of these SRs?

Status: [] Open

Response: *The "real" frequency of these SRs is monthly for most components unless modifications are made, or if a fire (i.e., smoke) or painting occurs in the vicinity.*

- iv. Condition F was relocated above Condition G consistent with the ITS Writer's Guide. This is an ITS Category (iii) change.

[ITS62.iv]:

3.6Q88 This appears to be purely administrative but why is it categorized as technical?

Status: [] Open

Response: *NUREG-1431 LCO 3.6.6A is technically broke with respect to Condition F regardless of the changes being proposed. Condition F requires a plant shutdown if Condition C is not met. Based on the ITS Writer's Guide, Condition F should immediately follow Condition C.*

- v. Incorporation of approved Traveller NRC-03, C.9, Revision 1. Not checked .

- vi. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including

providing consistency with current Ginna Station TS bases.

- b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections

[ITS62.vi.a and b]:

3.6Q89 There is an insert at the end of the second paragraph of Background for CS System which is not understood for "additional heat removal"?

Status:[] Open

Response: *During the recirculation mode following an accident, the CS system is in pull-stop unless CNMT pressure reaches 37 psig at which time CS is reinstated. Containment heat removal during recirculation is normally provided by the CRFCs and the cooling via the sump (i.e., through the RHR system).*

3.6Q90 It appears that the last paragraph of insert 3.6.6.3 should be relocated to the end of the paragraph into which it is inserted?

Status:[] Open

Response: *Attachment C of the submittal shows that the last paragraph of Insert 3.6.6.3 is separate from the last sentence contained in the inserted paragraph.*

3.6Q91 What are the other Ginna containment ventilation and air conditioning systems noted on Page B 3.6-65?

Status:[] Open

Response: *There are essentially five systems which can be used to provide cooling to various points within containment during MODES 1-4. These systems are listed below and discussed in UFSAR Section 9.4.1:*

- a. *CRFC units.*
- b. *Control Rod Drive Mechanism Cooling System*
- c. *Reactor Compartment Cooling System*
- d. *Containment Mini-Purge System*
- e. *Penetration Cooling System*

3.6Q92 In the Applicable Safety Analyses, there are three deleted paragraphs which should be specifically explained.

Status:[] Open

Response: *The first paragraph of concern deletes text related to worst case single active failure as discussed in the response to 3.6Q73. [phone for inadvertent spray]. The third paragraph of concern is deleted in its entirety. This paragraph specifies that containment cooling train performances under varying accident conditions is specified in the UFSAR. This level of detail is not in the current Ginna Station UFSAR and is instead retained within various analyses and other documents which are not generally available to outside parties (e.g., Westinghouse proprietary analyses).*

3.6Q93 Insert 3.6.6.11 is missing.

Status:[] Open

Response: *The requested insert 3.6.6.11 is being provided.*

3.6Q94 Questions to the LCO parts of 3.6.6 have asked for sketches to show the relationship and dependencies of these three systems. The improved TS LCO basis at this point does not appear to agree with

the existing TS as is explained and interpreted by CTS 13.xv.
Status:[] Open
Response: *To be discussed at the meeting.*

3.6Q95 The questions to the LCO part of 3.6.6 need to be addressed, discussed and resolved before any continuing detailed review of the BASES LCO, ACTIONS and SRs section can occur.

Status:[] Open
Response: *To be discussed at the meeting.*

vii. Incorporation of approved Traveller WOG-23, C.6. Not checked

viii. The Completion Time limit of "10 days from the discovery of failure to meet the LCO" was not added to the new specification since Ginna Station currently does not have this requirement. The intent of adding this limit to the Completion Time is to prevent a plant from continuously being in the LCO without ever meeting the full system requirements. This abuse of the LCO is best handled under plant procedures since the addition of this limit to the Completion Time column creates confusion among licensed personnel. Providing this limit can still result in LCO abuse since the systems can all be declared OPERABLE for only a few minutes and then the LCO immediately entered again. Sufficient NRC guidance already exists with respect to extensive use of LCO time (e.g., Ref. 26). In addition, the Maintenance Rule (10 CFR 50.65) requires monitoring of equipment performance. Finally, a review of Ginna Station plant records indicates that the systems covered by this LCO were out of service a total of 1017 hours from June 1990 to July 1994 (or < 4% of the time in which the plant was above Cold Shutdown) which demonstrates that this limit is unnecessary.

[ITS62.viii]:

3.6Q96 It is acceptable to not add this but the justification raises a few questions. First, by deleting this we do not want to push the "abuse" of this situation further underground into plant procedures. The NUREG-1431 attempted to deal head-on with this "abuse" formerly called the "flip-flop" between Conditions. Have all places in the improved TS where this is used, been deleted? Has Completion Times 1.3 be changed? These are generic questions outside of this specific deletion in this section. Also, the 1017 hours out of service seems high for these systems alone.

Status:[] Open

Response: *All uses of this Completion Time limit have been removed from the specifications including the Completion Time 1.3 examples. There is also a Westinghouse proposed Traveller in the system with respect to this. The 1017 hours is the total time in which any one of the four CRFC units, either CS train, or the spray additive tank was inoperable over a 4 year window. Since most of these components are allowed to be inoperable for up to 7 days (or 168 hours), this total is not considered excessive.*

ix. SR 3.6.6.A.3 was not added to the new specifications. This SR requires verification of a minimum SW flow rate through the fan coolers. This process variable is not used or credited in the DBA or transient analyses. The current Ginna Station TS do not contain

this surveillance. In addition, no other component supplied cooling water by SW (e.g., DGs, CCW) has any flow rate verification surveillance. This is an ITS Category (i) change.

[ITS62.ix]:
3.6Q97

The correct SW flow rate is important to keep the CRFC units OPERABLE since two units are always running during normal operations to maintain the containment air temperature within the new LCO 3.6.5 limits. These limits were not previously in the existing TS. This verification is important because the SW flow is apparently organized at the pump level and not the loop level to these units. Please add this SR.

Status: []
Response:

Rejected

The CRFC units use SW to remove containment heat during both normal operation and accident conditions. This heat removal capability is based on both SW flow and temperature. That is, with higher SW temperatures, you require higher flowrates to maintain heat removal requirements. The accident analyses have been performed assuming the highest SW temperatures to create worst case conditions. Therefore, requiring these SW flowrates during winter months is excessively conservative due to the low water temperatures which exist. In addition, during winter months all four CRFC units have flow through them with a common discharge AOV throttling flow. In order to perform this test, the common AOV would have to be opened causing a temperature transient within containment. Similar tests of SW supply to the CCW heat exchangers and the diesel generators are not required by NUREG-1431. An industry traveller is currently in the system to delete this requirement. [This response was changed as a result of meetings the week of 10/9/95. See comment #131]

63. ITS 3.6.7

- i. This section and associated bases were not added since they were relocated to LCO 3.6.6 as discussed above. This is an ITS Category (i) change. ACCEPTABLE

64. ITS 3.6.8

- i. The Ginna Station containment design is a large dry structure typical for a single unit PWR. Therefore, all bases and headers associated with subatmospheric, ice condenser and dual containment structures were deleted. In addition, the LCO was renumbered since LCO 3.6.7 was relocated to LCO 3.6.6. This is an ITS Category (iv) change. ACCEPTABLE
- ii. Incorporation of approved Traveller BWR-06, C.5. ACCEPTABLE
- iii. Incorporation of approved Traveller WOG-11, C.5. Not checked
- iv. Incorporation of approved Traveller WOG-11, C.1. Not checked
- v. SR 3.6.8.1, SR 3.6.8.2, and SR 3.6.8.3 were not added since the current Ginna Station TS do not contain a requirement for the hydrogen recombiners or these surveillances. As described in the new bases for this section, the hydrogen recombiners installed at

GINNA Station are inside containment and are designed around the use of a combustion chamber to control hydrogen generation. Performing a functional test would most likely require an evacuation of containment for safety reasons with little benefit. Instead, RG&E proposes to perform a CHANNEL CALIBRATION of each hydrogen recombiner actuation and control channel every 24 months to ensure that each hydrogen recombiner will provide the correct hydrogen and oxygen mixture in the combustion chamber. In addition, the blower fan for each hydrogen recombiner will be operated for ≥ 5 minutes every 24 months. This is an ITS Category (i) change.

[ITS64.v]:
3.6Q98

Operating the fan is a good start but there should also be a minimum amount of testing that is possible to verify that 1) supply hydrogen and oxygen gets to the unit to permit combustion; 2) the power supply is independently redundant; 3) the unit can flash or spark without hydrogen/oxygen present; 4) that moisture or other by-products do not foul operation of the unit; and etc. The insert 3.6.7.5 implies only the fan is needed to oxidize the hydrogen within containment. This does not appear to be correct. Revised this new SR 3.6.7.1

Status: []
Response:

Open
Each of the above issues is addressed separately below (note, a sketch of the recombiner and its control logic are attached):

1. A test gas (e.g., nitrogen) will be supplied to the recombiner every 24 months to verify that no plugging of these lines has occurred.
2. Verification that the power supply is independently redundant is not required since each hydrogen recombiner is permanently powered from opposite electrical trains.
3. A sightglass exists on the units such that the ignitor can be actuated and a verification made that sparks were generated. This verification will be performed every 24 months.
4. The fouling of the unit will be addressed by operating the blower fan, stroking all of the control valves in the control logic, and performing calibrations of this control logic every 4 months.

Essentially, all components of the hydrogen recombiner will be tested separately to ensure that collectively, the recombiner will remain OPERABLE. The recombiner was last functionally tested in 1979 which resulted in 300°F air being discharged into a mostly evacuated containment. Comment #45 has been opened to address these testing requirements.

3.6Q99 What is wrong or difficult with performing a physical and visual inspection of the recombiner as is required in SR 3.6.8.2?

Status: []
Response:

Open
RG&E agrees to add NUREG-1431 SR 3.6.8.2 to perform a physical and visual inspection of the recombiner every 24 months. Comment #45 has been opened to address this.

3.6Q100 Since these are the only hydrogen control units at the plant why isn't there a verification of operation every 184 days? Such as operating the blower fan or performing a channel check or calibration.

Status: [] Open

Response: *As discussed in the response to 3.6Q101, the NRC has approved the use of a hydrogen purge system. Therefore, RG&E does not believe that additional testing is required.*

vi. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including discussion concerning the design of the hydrogen recombiners.

b. Ginna Station has an alternate hydrogen purge system as described in UFSAR Section 6.2.5.2.2 and accepted by the NRC in Reference 27.

[ITS64.vi.a and b]:

3.6Q101 There is some confusion on system names. Does this mean Ginna has a hydrogen purge system or does this mean Ginna has an alternate hydrogen control system. The text at the bottom of page B 3.6-115 has deleted reference to the hydrogen purge system. In order to have a Condition B to this LCO, an alternate hydrogen control system must be acceptable to the NRC staff. Page B 3.6-117 implies this is the Mini-Purge System. Is this correct?

Status: [] Open

Response: *Ginna Station utilizes the Mini-Purge system as an alternate hydrogen purge system (see UFSAR Section 6.2.5.2). This technique was accepted by the NRC as a backup hydrogen purge system per letter from D.M. Crutchfield, NRC, to L.D. White, RG&E, dated July 7, 1980 (attached). The actual hydrogen purge system accepted by the NRC was the shutdown purge system since the Mini-Purge System was not installed until the mid-1980s. However, the Mini-Purge System is now the preferred path.*

3.6Q102 The background to the BASES in the second paragraph is revised to imply hydrogen will be discharged to the environment during normal operation by use of the mini-purge system and during accident conditions as the alternate to the recombiners. See BASES insert 3.6.7.3 which proposes evacuations of the general public? Based on this information there is no reason to have a Condition B for Ginna.

Status: [] Open

Response: *See response to 3.6Q100. The use of Condition B due to the purging capability of the Mini-Purge System is considered acceptable since the NUREG-1431 bases reference the use of a Hydrogen Purge System.*

3.6Q103 The BASES insert 3.6.7.4 is misleading. It is stated that recombiners must be placed into operation before the 4.1 v/o limit is reached. Why imply that the limit could be exceeded sooner and not reduced the Completion Times to match this analyzed state? Also implying operation of the recombiners at or near the 6 v/o limit does not seem prudent. There is an implied dependence on venting excess hydrogen which suggests the recombiners do not have 100% redundant capability.

Status: [] Open

Response: *As discussed in UFSAR Section 6.2.5.2, two analyses were performed with respect to hydrogen generation within containment. Both*

analyses demonstrate that the hydrogen flammability limit of 6.0 v/o is not reached until at least 31 days following an accident. There is no dependence of venting excess hydrogen at any time below this 6.0 v/o limit since hydrogen is physically piped into containment during use of the recombiners to ensure that it is appropriately burned in the combustion chamber. The rate at which hydrogen is added is dependant upon the containment hydrogen concentration. RG&E proposes to revise the last sentence of Insert 3.6.7.4 to read "Operation of the hydrogen recombiners ensures that a concentration of 6.0 v/o would not be reached inside containment which could result in an overpressurization given an ignition source." Comment #46 has been opened to address this.

65. ITS 3.6.9

- i. This section and associated bases were not added. The Hydrogen Mixing System as defined in the bases is used to ensure that containment atmosphere is uniformly mixed. Ginna Station does not have a Hydrogen Mixing System and instead uses the Containment Recirculation Fan Cooling System for this function (LCO 3.6.6). Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

[ITS65.i]:
3.6Q104

Why is the CRFC system not used as an alternate hydrogen control system?

Status: []
Response:

Open
The CRFC system by itself cannot remove or purge hydrogen from containment since it only cools and recirculates air within the containment. The Mini-Purge System must also be in service to act as a hydrogen purge system. Since the CRFC system is already required by LCO 3.6.6, RG&E does not believe that this LCO is required.

3.6Q105 Has the NRC staff provided an evaluation/acceptance of the hydrogen control methods and alternatives at Ginna?

Status: []
Response:

Open
See response to 3.6Q101.

3.6Q106 Wouldn't the use of air circulation within containment preclude the formation of hydrogen pockets with potential concentrations above the 6 v/o limit?

Status: []
Response:

Open
The use of air circulation would assist in the reducing the formation of air pockets but no analyses have been performed to justify this conclusion.

66. ITS 3.6.10

- i. This section and associated bases were not added. The Hydrogen Ignition System as defined in the bases is used to control hydrogen levels within containment post accident. Ginna Station does not have a Hydrogen Ignition System and instead uses the Hydrogen Recombiner System for this function (LCO 3.6.7). Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS

Category (i) change.

[ITS66.i]:
3.6Q107

It is acceptable not to add this LCO due to the TSIP conversion guidelines prohibiting plant modifications to implement the NUREG-1431.

Status: [] Closed
Response: N/A

67. ITS 3.6.11

- i. This section and associated bases were not added since they were relocated to LCO 3.6.6 as discussed above. This is an ITS Category (i) change.

68. ITS 3.6.12

- i. This section and associated bases were not added. The function of the containment vacuum relief valves as defined in the bases is to ensure that containment is protected against negative pressure. Ginna Station does not have containment vacuum relief valves. Protection against negative pressure is provided by LCO 3.6.4, "Containment Pressure." Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

ITS68.i:
3.6Q108

Please explain how Ginna does not need to have a Containment Vacuum Relief System. The existence of a pressure limitation does not preclude the need to analyze for a potential situation such as an inadvertent actuation of the containment spray system. Has this analysis been performed? Is there sufficient margin between the negative pressure created and the structural limit of the containment liner.

Status: [] Open
Response: *To be discussed at the meeting.*

69. ITS 3.6.13

- i. This section and associated bases were not added. The Shield Building Air Cleanup System (SBACS) as defined in the bases is used to ensure that radioactive materials which leak from containment following a DBA is adequately filtered and absorbed. Ginna Station does not have a SBACS since the leakage through the containment liner is controlled by LCO 3.6.1, "Containment." Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

70. ITS 3.6.14

- i. This section and associated bases were not added. The Air Return System as defined in the bases is only used at Ice Condenser designs which does not apply to Ginna Station. Therefore, this requirement is not relevant to the Ginna

Station design. This is an ITS Category (i) change.

71. ITS 3.6.15

- i. This section and associated bases were not added. An ice bed as defined in the bases is only used at Ice Condenser designs which does not apply to Ginna Station. Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

72. ITS 3.6.16

- i. This section and associated bases were not added. Ice condenser doors as defined in the bases are only used at Ice Condenser designs which does not apply to Ginna Station. Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

73. ITS 3.6.17

- i. This section and associated bases were not added. A divider barrier as defined in the bases is only used at Ice Condenser designs which does not apply to Ginna Station. Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

74. ITS 3.6.18

- i. This section and associated bases were not added. Containment recirculation drains as defined in the bases are only used at Ice Condenser designs which does not apply to Ginna Station. Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

75. ITS 3.6.19

- i. This section and associated bases were not added. A Shield Building as defined in the bases is used to ensure that radioactive materials which leak from containment following a DBA are adequately filtered and absorbed. Ginna Station does not have a Shield Building since the leakage through the containment liner is controlled by LCO 3.6.1, "Containment." Therefore, this requirement is not relevant to the Ginna Station design. This is an ITS Category (i) change.

Section 3.7 Current TS

13. Technical Specification 3.3

- i. TS 3.3.1.1.b and 3.3.1.3 - LCO 3.5.1 Condition A was added which allo.....
 - .
 - .
 - .
- xv. TS 3.3.2.2 - This was revised to allow both post-accident charcoal

filter trains (including the CRFC units which supply them) to be inoperable for up to 72 hours if both containment spray (CS) trains are OPERABLE. This change provides consistency with the accident analyses which demonstrate that either two CS trains, one CS train and one post-accident charcoal filter train, or two post-accident charcoal filter trains are adequate to remove radioactive iodine from the containment atmosphere following a DBA (i.e., each CS train and post-accident charcoal filter train provides 50% of the required iodine removal requirements). However, two CS trains cannot be inoperable since at least one train must operate for containment pressure and temperature control. In addition, two CRFC units can now be removed from service for up to 7 days since the accident analyses only credit two of the four cooling units as being OPERABLE with respect to containment pressure and temperature control. Finally, with one or two CRFC units inoperable and not restored within 7 days, the plant has only 36 hours to reach MODE 5 versus 84 hours due to the importance of maintaining containment pressure and temperature control. These are Ginna TS Category (v.b.12) changes.

[CTS13.xv-L1]:

3.6Q109 The existing TS 3.3.2.2.f is missing from the CTS. Is this a typo or an error?

Status: [] Open

Response: *There is no TS 3.3.2.2.f in the CTS, nor any discussed in CTS change 13.xv.*

3.6Q110 Is this a new DBA analysis which has been performed since these CTS were issued as Amendment 24?

Status: [] Open

Response: *No, the same DBA analysis which was performed in support of Amendment #24 is used as the basis for this change. As a matter-of-fact, the CTS bases on page 3.3-12 (identified as the "unidentified page" below) contain much of the information used to justify this proposed change. The CTS are also "broke" in that it does not allow more than one CRFC to be inoperable at a time. However, there are two CRFCs per electrical train (actually the same 480V bus), which CTS 3.7.2.2.c allows to be removed from service for up to 1 hour.*

3.6Q111 The unidentified page following TS 3.3.2.2.e contains three combinations of systems to meet the DBA. Where is Ginna's commitment made to hence forth not rely on "(2) two CRFC units and two post-accident charcoal filters" to satisfy the DBA accident analysis.

Status: [] Open

Response: *ITS LCO 3.6.6 requires all 4 CRFC units, both CS trains, and the spray additive tank to be OPERABLE. The various combinations of equipment which is allowed to inoperable at any one time does not allow the use of only the "two CRFC units and two post-accident charcoal filters" in item (2).*

3.6Q112 Please show the configuration and number of filters in the sketch requested in ITS62.iii,item #2.

Status: [] Open

Response: *The requested sketches are being included.*

16. Technical Specification 3.6

- i. TS 3.6.1 - The text allowing closed containment isolation valves to be opened on an intermittent basis under administrative controls was relocated to a LCO Note consistent with NUREG-1431. This is a Ginna TS Category (v.c) change. Accepted as technically equivalent and is administrative in nature; therefore, no SE mention needed.
- ii. TS 3.6.2 - The Applicability for maintaining containment pressure within limits was revised from reactor criticality to MODE 4. This change is necessary to provide consistency with the requirements for containment integrity (i.e., LCO 3.6.1) since exceeding these pressure limits could result in a overpressure of containment if an accident were to occur. This is a Ginna TS Category (iv.a) change.

[CTS 16.ii-M1]:

3.6Q113 This is acceptable.

Status: Closed

Response: N/A

[CTS16.ii-L1]:

3.6Q114 Explain how containment pressure is affected by the limiting size and function of Mini-Purge System.

Status: Open

Response: See response to 3.6Q57

3.6Q115 How long does it take to open the isolation valves to return to atmospheric pressure?

Status: Open

Response: See response to 3.6Q58

3.6Q116 Is this pathway filtered?

Status: Open

Response: See response to 3.6Q59

3.6Q117 How long does it take to exchange containment air volume?

Status: Open

Response: See response to 3.6Q60

3.6Q118 Why 24 hours and not a Completion Time of 1, 2, 4 or 8 hours? Remember the original basis for 1 hour is consistency with the loss of the new containment operability per ITS 3.6.1!

Status: Open

Response: See response to 3.6Q61

- iii. TS 3.6.3 - The title for this LCO was revised from containment isolation "boundary" to "barrier" which provides greater consistency with the bases for NUREG-1431. In addition, three new requirements were added. The first requires that a penetration with both containment barriers inoperable be isolated within 1 hour versus 4 hours. This change provides consistency with TS 3.6.1 since containment integrity is potentially violated. As such, verification of continued acceptable containment leakage must be initiated immediately if both barriers are declared inoperable. In

addition, new requirements with respect to an inoperable airlock (including the use of an airlock with an inoperable door or interlock mechanism) and containment mini-purge penetrations with isolation valves that exceed their leakage rate acceptance criteria were added. The new requirement for the air locks specifies that an inoperable airlock door (including an inoperable interlock mechanism) must be isolated within 1 hour and locked closed within 24 hours. However, a dedicated individual can be used in the case of an inoperable interlock mechanism to allow entry and exit through the airlock. The new specification provides specific Required Actions in the event that current Ginna Station TS 4.4.2.4.c is exceeded. The new requirement for the mini-purge penetrations specifies that the affected penetration must be isolated within 24 hours if an isolation valve exceeds its leakage limit. These new requirements provide added assurance that penetrations which can provide direct access to the outside environment are addressed quickly when their isolation barriers become inoperable. This is a Ginna TS Category (iv.a) change.

[CTS16.iii-L1]: Also see [ITS59.ii]

3.6Q119 See 3.6Q

Status: [] Rejected, 7-14-95

[] Open, 7-29-95

Response: *This item was originally rejected but it was reopened following a telecon on 7-27-95. The name change from containment isolation "boundaries" to "barriers" is accepted for the title of LCO 3.6.3; however, internally within the LCO each name change must be discussed.
To be discussed at the meeting.*

[CTS16.iii-M1]:

3.6Q120 The change to restore containment integrity within one hour rather than 4 hours per existing TS 3.6.3 is acceptable.

Status: [] Closed

Response: N/A

3.6Q121 What does 4th sentence of #16.iii mean?

Status: [] Open

Response: *This sentence refers to new Required Action B.2 which requires evaluation of containment integrity within 24 hours after two containment isolation valves for the same penetration are declared inoperable.*

3.6Q122 What does 8th sentence of #16.iii mean? There are no limits stated; so how is this TS exceeded?

Status: [] Open

Response: *This is a typographical error in Attachment A to the submittal in that it should read CTS "4.4.2.3.c" and not "4.4.2.4.c" at the end of this sentence. Comment #47 has been opened to correct this.*

3.6Q123 See ITS 3.6.3 #59.iv for other comments on improved TS.

Status: [] Open

Response: *To be discussed at the meeting.*

[CTS16.iii-L2]:

3.6Q124 The relaxations for LCO 3.6.2 Conditions A and C have not been justified.

Status: Open

Response: *Ginna Station currently treats air lock doors under CTS 3.6.3 since there is no specific air lock LCO. Since each door in both airlocks contains redundant testable seals; the OPERABILITY of either door and its two associated seals meets the requirements of CTS 3.6.3. Therefore, if an airlock door is inoperable, CTS 3.6.3 has no Required Actions (although the Ginna Station operating practices would require restoring the door to OPERABLE status within a prudent time frame). If both doors were inoperable, then CTS 3.6.3 requires 4 hours to isolate the airlock if containment leakage limits are met per CTS 3.6.1; otherwise only 1 hour is allowed. Therefore, the ITS LCO 3.6.2 Conditions A (which corresponds to one inoperable airlock door) and C (which corresponds to two inoperable airlock doors) are more restrictive than CTS requirements.*

[CTS16.iii-L3]:

3.6Q125 The relaxation for new LCO 3.6.2, Condition B has been accepted.

Status: Closed

Response: N/A

[CTS16.iii-L4]:

3.6Q126 The relaxation for ITS LCO 3.6.2 Actions Note #1 has not been justified.

Status: Open

Response: *As discussed in the response to 3.6Q124, Ginna Station currently has no air lock requirements and considers air lock doors under LCO 3.6.3. If an air lock door were inoperable, no further required action is necessary in the CTS. If access through the OPERABLE door in the affected air lock were required, then CTS 3.6.1 would apply which allows 1 hour to restore containment integrity. However, the remaining air lock would be used to the extent practical under this circumstance. Therefore, the implementation of this Action Note is consistent with current practices and requirements.*

[CTS16.iii-L5]:

3.6Q127 Is there a need for any routine access to containment? What is it?

Status: Open

Response: *Ginna Station personnel normally enter containment at least once per month to perform TS required surveillances. These include HEPA and charcoal filter testing of the CRFC units (plus other non-TS ventilation systems) and the 15 minute run of the post-accident charcoal filters. The accumulators are also sampled monthly with ASME valve tests performed quarterly.*

3.6Q128 The relaxation for Condition A, Required Actions, Note #2 has not been justified.

Status: Open

Response: *As discussed in the response to 3.6Q124, Ginna Station currently has no air lock requirements and considers air lock doors under LCO 3.6.3. If both air lock doors were inoperable, then CTS 3.6.1 must be met with respect to containment integrity. If containment integrity is maintained, then the airlock would essentially be locked closed to prevent further use. If access through the*

affected air lock were required, then CTS 3.6.1 would apply which allows 1 hour to restore containment integrity. However, the remaining air lock would be used to the extent practical under this circumstance. Therefore, the implementation of this Action Note is consistent with current practices and requirements.

[CTS16.iii-L6]:

3.6Q129 The relaxation for LCO 3.6.3, Condition A has not been justified.
Status: [] Open
Response: *To be discussed at the meeting.*

[CTS16.iii-L7]:

3.6Q130 The improved TS LCO 3.6.3 Condition E (new Condition C) is a relaxation which needs justification for both shutdown purge and mini-purge valves.

Status: [] Open

Response: *ITS LCO 3.6.3 Condition C is a more restrictive change with respect to CTS and the mini-purge valves. CTS 4.4.2.3.c specifies leakage limits for the mini-purge valves and then states that if these limits are not met, "an engineering evaluation shall be performed and plans for corrective action developed." ITS LCO 3.6.3 Condition C requires isolation of the penetration within 24 hours and verification that it is closed once every 31 days. See Attachment A, section D, item 31.iv. Condition C does not apply to the shutdown purge system.*

3.6Q131 The original intent of this Condition was for large purge valves which are inherently more difficult to restore operable. Why should the mini-purge valves be given this large Completion Time and not held to 4 hours?

Status: [] Open

Response: *First, Ginna Station currently does not have any isolation requirements with respect to excessive leakage as discussed in the response to 3.6Q130. Second, the mini-purge system is the primary system available to maintain containment pressure within limits of ITS LCO 3.6.4. Requiring isolation of this penetration in a short period of time would reduce the ability to reduce containment pressure when required due to the upper limit of 1.0 psig.*

- iv. TS 3.6.3 - The use of a closed system to isolate an inoperable containment isolation barrier was added to this specification. Consequently, a closed system which must be OPERABLE to meet this specification can be used to isolate a failed isolation barrier. Also, isolation devices located outside containment that were used to isolate a failed containment isolation valve are required to be verified closed once every 31 days. For isolation devices inside containment, they must be verified closed upon entry into MODE 4 from MODE 5 if it has not been performed within the last 92 days. These are Ginna TS Category (v.b.22) changes.

[CTS16.iv-L1]:

3.6Q132 The use of a closed system to isolate an inoperable containment isolation barrier is not accepted as proposed. The merging of this new isolation method into Condition A is confusing and requires new information. Please refer to ITS 59.iv for more questions.

Status: [] Open
Response: *To be discussed at the meeting.*

[CTS16.iv-M1]:
3.6Q133 It is acceptable to reverify the isolation of a penetration at different intervals depending whether the isolation device is located inside or outside of containment.

Status: [] Closed
Response: *N/A*

- v. TS 3.6.5 - This was relocated to the bases for ITS 3.6.3 since it does not meet any of the four criteria and does not specify any Required Actions. Operation of the Mini-Purge System is performed under procedures such that its use is strictly controlled. Placing this information in the bases also provides similar control under 10 CFR 50.59 (i.e., the Bases Control Program). This is a Ginna TS Category (iii) change.

[CTS16.v-RI1]:
3.6Q134 TS 3.6.5 seems to be the same as SR 3.6.3.2 so this should be a Category (i) change. The text needs to be in the BASES but should be in the BASES for describing the purpose of SR 3.6.3.2.

Status: [] Open
Response: *See response to 3.6Q41.*

- vi. TS 3.6 - A new requirement was added which specifies that the average containment air temperature shall be $\leq 120^{\circ}\text{F}$ above MODE 5. This temperature limit is necessary to ensure that the resulting containment temperature following a DBA is within the assumptions used for environmental qualification of components within containment. If the average containment air temperature is $> 120^{\circ}\text{F}$, it must be restored within 24 hours. This is a Ginna TS Category (iv.a) change.

[CTS16.vi-M1]:
3.6Q135 The addition of the new LCO is acceptable; however, the 24 hours to restore OPERABLE needs review as is noted in ITS61.ii.

Status: [] Open
Response: *See responses to 3.6Q68 through 3.6Q71.*

- vii. TS 3.6 - A new requirement was added which requires the hydrogen recombiners to be OPERABLE in MODES 1 and 2. The hydrogen recombiners are assumed in the accident analyses to be used to prevent a hydrogen explosion within containment that could overpressurize the containment structure. The new LCO allows 30 days to restore an inoperable recombiner and 7 days to restore two inoperable recombiners if the Mini-Purge System is OPERABLE. In addition, the plant can enter MODES 1 and 2 with an inoperable hydrogen recombiner. This is a Ginna TS Category (iv.a) change.

[CTS16.vii-M1]:
3.6Q136 It is acceptable to add the new LCO 3.6.7 for the Hydrogen Recombiner.

Status: [] Closed
Response: *N/A*

3.6Q137 The Applicability of this LCO is MODE 1 and 2 but CTS 16.viii state applicable DBAs are assumed to occur also in MODE 3. Please explain why LCO is also not applicable in MODE 3.

Status:[] Open

Response: *The reason that the hydrogen recombiners are not required below MODE 2 is explained in the Applicability bases of ITS LCO 3.6.7. Essentially, the hydrogen production rate following a DBA is lower in MODES 3 and 4 while the probability of an accident requiring the recombiners is very low in all MODES below MODE 2.*

3.6Q138 The existing TS had no mention of recombiners, but the NUREG-1431 guidance assumes there is at least one if not two alternate methods of hydrogen control. There appears to be none at Ginna except the release of hydrogen directly to the atmosphere under normal and accident conditions. Based on this, only Condition A is acceptable and Condition B can not be allowed.

Status:[] Open

Response: *Please see response to 3.6Q101*

- x. TS 3.6.1.b and TS 3.6.1.c - The requirement describing the specific applicability for containment integrity was not added. No screening criteria apply for this requirement since containment integrity is not assumed in the refueling safety analysis. The fuel handling accident inside containment analysis (UFSAR 15.7.3.3) takes no credit for isolation of the containment, containment integrity, nor effluent filtration prior to release. The requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. Boron concentration changes in MODE 6 and Required Actions to suspend positive reactivity additions is provided in new LCO 3.9.1. This is a Ginna TS Category (iii) change.

[CTS 16.x-RI1?]:

3.6Q139 This is proposed to be relocated to refueling but the questions on why there is no Containment Integrity for MODE 6 still remain.

Status:[] Open

Response: *RG&E has proposed to relocate the MODE 6 containment requirements to the TRM as discussed in Attachment A, Section C, item 107.i. [This response was revised as a result of 11/16/95 Appeal meeting. See comment #221.]*

28. Technical Specification 4.1

- i. The following changes were made to TS 4.1.1 or Table 4.1-1:
 - k. Table 4.1-1, Functional Unit #25 - The calibration and testing requirements for the containment pressure narrow range transmitter were not added to the new specifications. This instrument is not used or credited in any DBA or transient analysis. This instrument is only used to verify that containment pressure remains ≤ 1.0 psig and ≥ -2.0 psig during normal operation. These items were relocated to the RM. This is a Ginna TS Category (iii) change.

[CTS28.i.k-R01]:

3.6Q140 Further explain this justification for relocation. Is this the instrument which is used to verify SR 3.6.4.1? If it is, then this parameter is the assumed initial ambient pressure for determining the peak containment pressure in the DBA analysis. Likewise, peak negative pressure for inadvertent containment spray would also be based on this instrument.

Status: [] Open

Response: *Change D.28.i.k is a typographical error and should be deleted. The "Remarks" column for this surveillance specifically states that the "narrow range containment pressure (-3.0, +3 psig) excluded." Therefore, this discussion is not relevant to the actual surveillance requirements for SR 3.6.4.1. Comment #47 has been opened to delete this item.*

ii. The following changes were made to TS 4.1.2 or Table 4.1-2:

- e. Table 4.1-2, Functional Unit #13 was revised per SR 3.6.6.8 to require verification of the spray additive tank NaOH concentration once every 184 days instead of monthly. This change is acceptable since the spray additive tank is normally maintained isolated at power such that changes to the NaOH concentration or level are not expected. This is a Ginna TS Category (v.b.30) change.

[CTS28.ii.e-L1]:

3.6Q141 Please describe what makeup water volume pathways to the spray additive tank exist and how is it only through which valves that are locked, isolated or under administrative control. This is to support the basis that the tank is isolated at power.

Status: [] Open

Response: *The only makeup water volume pathway to the spray additive tank is from the Primary Water Treatment System. The use of this pathway requires opening of 5 normally closed, in-series manual valves. One of these valves is maintained locked closed. The tank drain line contains a locked closed manual valve.*

i. The following new requirements were added to Table 4.1-2 (Ginna TS Category (iv.a) changes):

- 10. SR 3.6.5.1 - requires verification every 24 hours that containment average air temperature is $\leq 120^{\circ}\text{F}$.

[CTS28.ii.i.10-M1]:

3.6Q142 This is acceptable.

Status: [] Closed

Response: N/A

- 11. SR 3.6.6.7 - requires verification every 184 days that the spray additive tank volume is ≥ 4500 gallons.

[CTS28.ii.i.11-M1]

3.6Q143 Please describe what makeup water volume pathways from the spray additive tank exist and how it could only be through which valves that are locked, isolated or under administrative control that water could be inadvertently released. This is to support the basis for the 184 days interval during which the tank will not be inadvertently drained.

Status: [] Open

Response: *Please see response to 3.6Q141.*

31. Technical Specification 4.4

- i. TS 4.4.4 - The requirements for the tendon stress surveillances were not added. The level of detail is relocated to the Pre-stressed Concrete Containment Tendon Surveillance Program described in new Specification 5.5.6 and a more generic program description is provided. This is a Ginna TS Category (iii) change.

[CTS31.i-R01]:

3.6Q144 Ginna has made changes to the applicable Section 5.5.6 which need resolution prior to approval of this SE.

Status: [] Open

Response: *To be discussed at the meeting.*

- ii. TS 4.4.3 - The requirements for the testing of the portion of the RHR system in the recirculation configuration were not added. The level of detail is relocated to the Primary Coolant Sources Outside Containment Program described in new Specification 5.5.2 and a more generic program description is provided. This is a Ginna TS Category (iii) change.

[CTS31.ii-R01]:

3.6Q145 Ginna has made changes to the applicable Section 5.5.2 which need resolution prior to approval of this SE.

Status: [] Open

Response: *To be discussed at the meeting.*

- iii. TS 4.4.1 (except definition for L_a), 4.4.2.1, 4.4.2.2, and 4.4.2.4 - These were not added to the new specifications since this information is contained in 10 CFR 50, Appendix J and does not need to be retained within technical specifications. SRs 3.6.1.1 and 3.6.1.2 provide for the necessary relation from technical specifications to Appendix J. These are Ginna TS Category (ii) changes.

[CTS31.iii-R01]:

3.6Q146 The contents of existing TS 4.4.2.4.b needs to be added to the Background BASES in item c which is proposed to be deleted. See ITS57.iv.

Status: [] Open

Response: *See response to 3.6Q12*

- iv. TS 4.4.2.3.a and 4.4.2.3.b - These were revised to require that if the allowed 10 CFR 50, Appendix J leakage limits are exceeded, they must be restored within 1 hour versus 48 hours consistent with LCO

3.6.1. However, the leakage limit of $< 0.6 L_0$ was revised to be consistent with the new Appendix J rule and implementation guidance (i.e., the leakage limit is $< 0.6 L_0$ on a maximum pathway leakage rate basis prior to entering MODE 4 for the first time following each refueling outage and $< 0.6 L_0$ on a minimum pathway leakage rate basis for all other time periods). This is a Ginna TS Category (v.a) change.

[CTS31.iv-L1]:

3.6Q147 The existing TS requirement is for penetrations and leakage paths under Type B and Type C testing. The "48 hours" to restore has been relaxed to "indefinite" by LCOs 3.6.2 and 3.6.3.

Status:[] Open

Response: *The 48 hours provided in CTS 4.4.2.3.b is the Completion Time for CTS 4.4.2.3.a which requires repairs if the "total leakage from all penetrations and isolation boundaries exceeds $0.6L_0$." 10 CFR 50, Appendix J specifies this same $0.6L_0$ acceptance criteria for Type B and C penetrations. Therefore, the Completion Time in CTS 4.4.2.3.b is in excess of that for CTS 3.6.1 and ITS LCO 3.6.1 which requires restoration within 1 hour or initiation of shutdown. As such, this is not a relaxation but a change which provides consistency within both the CTS and ITS.*

3.6Q148 Where does this "minimum/maximum pathway leakage rate" come from? This sounds like another relaxation?

Status:[] Open

Response: *See response to 3.6Q6.*

- v. TS 4.4.2.4.c - A specified air lock leakage acceptance criteria of $\leq 0.05L_0$ when tested at $\geq P_0$ was added to the new specifications. This acceptance criteria is required to be retained within technical specifications by 10 CFR 50, Appendix J, Section III.D.2(iv) and is consistent with NUREG-1431 and current testing requirements. In addition, a new Surveillance was added to verify that only one door in each airlock can be opened at a time once every 24 months. This test is necessary to ensure that the OPERABILITY of the air locks, as defined in the new bases for LCO 3.6.2 is maintained. These are Ginna Category (iv.a) changes.

[CTS31.v-L1]:

3.6Q149 Existing TS 4.4.2.4.a says penetrations, like air locks, are tested per Appendix J. Appendix J says the air locks have to have a leakage rated in the improved TS. Existing TS are deficient. This also contradicts ITS 58.xi justification which says it was revised. This seems not a more restrictive but is equivalent unless there is a previous requirement. Needs better justification.

Status:[] Open

Response: *See response to 3.6Q32.*

3.6Q150 See ITS58.iv - As noted above this SR is important to ensuring operability of the air locks so it should have the same Frequency of 184 days as leakage testing unless the air lock door is not opened. Justify testing interval at 24 months rather than 184 days.

Status:[] Open

Response: Please see response to 3.6Q16 and 3.6Q17.

3.6Q151 Appendix J is less restrictive now than existing TS 4.4.2.4.c in that once an air lock door is opened then you have 72 hours rather than 48 hours to retest the door seal. Provide this less restrictive justification.

Status: [] Open

Response: See response to 3.6Q18.

- vi. TS 4.4.2.3.c - The requirement to perform an engineering evaluation if the mini-purge supply and exhaust lines isolation valve leakage exceeds $0.05 L_a$ was revised to require isolation of the affected penetration within 24 hours. In addition, the affected penetration must be verified isolated once every 31 days if it is outside containment; or once every 92 days if it is inside containment. These changes provide direct guidance to operators which are consistent with NUREG-1431. This is a Ginna TS Category (v.c) change. This is acceptable. There is no need to include this in the SE since this temporary requirement is now made a permanent requirement.
- vii. TS 4.4.5.1 - Two new surveillances (SR 3.6.3.1 and SR 3.6.3.2) were added which require verification of the correct position of containment isolation barriers located outside containment once every 184 days and inside containment prior to entering MODE 4 from MODE 5 if it has not been performed within the previous 184 days. These surveillances ensure that the containment isolation barriers remain OPERABLE above MODE 5. These are Ginna TS Category (iv.a) changes.

[CTS31.vii-L1]:

3.6Q152 It is unclear how this is a more restrictive change. A tagging system is not 100% reliable nor is valve lock-out. This is the reason for this SR.

Status: [] Open

Response: This is a more restrictive change in that the CTS do not have any requirement to verify the correct position/alignment of containment isolation barriers. TS 4.4.5.1 only requires verification that containment isolation valves are OPERABLE in accordance with the IST Program. This program verifies that valves are capable of closing within their required time limits (e.g., 60 seconds), not that valves are in the correct position. The fact that Ginna Station currently performs position verification every 6 months and utilizes a tagging and locked valve program outside of TS requirements is not relevant to whether this is a more or less restrictive TS change.

3.6Q153 This SR contains a relaxation for those isolation devices located in high radiation areas. Justify this relaxation.

Status: [] Open

Response: This is a more restrictive change in that the CTS do not have any requirement to verify the correct position/alignment of containment isolation barriers. TS 4.4.5.1 only requires verification that containment isolation valves are OPERABLE in accordance with the IST Program. This program verifies that valves are capable of closing

within their required time limits (e.g., 60 seconds), not that valves are in the correct position.

3.6Q154 There should be maintenance/surveillance activities on-going in the plant continuously. It is for these reasons that the frequency of the SRs were kept at the intervals currently used in all active Westinghouse plants and all the other owner groups. During the development of the NUREG-1431, a lengthening of these 31 day intervals could not be justified, except for those located within containment to be at 92 days. The 184 days proposed is unacceptable. The periodic walkdowns are essential to verify equipment status and this is just another check which should be made in a consistent manner as them verifications required under the Required Actions of LCO 3.6.2 and LCO 3.6.3. As noted above in justification for CTS31.vi, this provides direct guidance to operators in conformance with the NUREG-1431 and all other plants.

Status:[]

Open

Response: See response to 3.6Q44.

- viii. TS 4.4.6.2 - The Surveillance Frequency for automatic containment isolation valves has been revised from 18 to 24 months (see Section D, item 1.xii). The response times for CIVs is discussed in the bases for new LCO 3.6.3. This is a Ginna TS Category (v.b.1) change.

[CTS31.viii-L1]:

3.6Q155 This relaxation from 18 to 24 months is being review by the Project Manager. This change is on hold until a decision is announced..

Status:[]

Open

Response: No response required by RG&E.

- ix. TS 4.4 - Two new Surveillances were added with respect to the hydrogen recombiners (SR 3.6.7.1 and SR 3.6.7.2). The first new Surveillance requires that the blower fan for the hydrogen recombiners be operated for ≥ 5 minutes once every 24 months. The second new Surveillance requires that a CHANNEL CALIBRATION be performed on the hydrogen recombiner actuation and control channels once every 24 months. The performance of these SRs ensures that the hydrogen recombiners are OPERABLE and capable of performing their post-accident function. These are Ginna TS Category (iv.a) changes.

[CTS31.ix-M1]:

3.6Q156 Please note that the comments state in ITS64.v equally apply here. Until there is agreement on the number and what are the contents of the new SRs it is difficult to justify them.

Status:[]

Open

Response: Please see responses to 3.6Q98 through 3.6Q100.

32. Technical Specification 4.5

- ii. TS 4.5.2.1 - This was revised to relocate all SI, RHR, and CS pump testing frequencies and discharge pressure requirements to the Inservice Testing program described in new Specification 5.5.8 consistent with the ITS. These are Ginna TS Category (iii) changes, respectively.

[CTS32.ii-LlorR01]:

3.6Q157 This is acceptable but isn't this a relaxation from once per month to the less frequent IST interval?

Status: [] Open

Response: *The IST program currently requires quarterly tests on this equipment such that this is actually a less restrictive change following implementation. The justification for this change is that ASME testing requirements only specify quarterly tests of pumps and valves as being adequate to demonstrate continued component OPERABILITY. The NRC has generically approved these testing frequencies via 10 CFR 50.55a and approval of the Ginna Station IST Program.*

- v. TS 4.5.2.3 - The requirements denoting the Frequency and conditions of the air filtration system tests were not added to the new specifications. This level of detail is relocated to the Ventilation Filter Testing Program described in new Specification 5.5.10. In addition, the remaining requirements were all relocated to the Administrative Controls section. These are Ginna TS Category (iii) and (i) changes, respectively.

[CTS32.v-R01]:

3.6Q158 This is acceptable.

Status: [] Open

Response: N/A

- vii. TS 4.5.1.2 - A new Surveillance (SR 3.6.6.1) was added to verify the correct position of each manual, power operated, and automatic valve in the CS flowpath that is not locked, sealed, or otherwise secured in position. This Surveillance ensures that the CS System is OPERABLE in accordance with the LCO. This is a Ginna TS Category (iv.a) change.

[CTS32.vii-M1]:

3.6Q159 The addition of SR 3.6.6.1 is acceptable.

Status: [] Open

Response: N/A

- viii. TS 4.5.1.2.b - The Frequency of performing the spray nozzle gas test was revised from once every 5 years to once every 10 years consistent with SR 3.6.6.14. The increased surveillance interval is considered acceptable due to the passive nature of the spray nozzles and previous acceptable results. This is a Ginna TS Category (v.b.36) change.

[CTS32.viii-L1]:

3.6Q160 The new SR 3.6.6.15 is acceptable.

Status: [] Open

Response: N/A

- ix. TS 4.5.2.3.5 - This was revised to only require actuation of the post-accident charcoal filter dampers from an actual or simulated SI signal once every 24 months to ensure that the system aligns itself correctly (SR 3.6.6.12). The post-accident charcoal filter dampers must still be opened at least once per 31 days to allow the system

to operate for ≥ 15 minutes. Consequently, only the frequency of the automatic alignment of the dampers is being revised to provide consistency with other specifications. This is a Ginna TS Category (v.b.37) change.

[CTS32.ix-L1]:

3.6Q161 This appears acceptable but explain if there is any difference between the CTS referring to isolation valves versus in the ITS reference to the dampers. Specifically in the Ginna terminology, are these dampers same as the isolation valves?

Status: [] Open

Response: *The phrase "post-accident charcoal filter isolation valves" in CTS 4.5.2.3.5 are the same devices as the dampers referenced in the ITS. As shown on the sketch provided in response to 3.6Q78, the dampers required to be tested by the CTS and ITS SR 3.6.6.12 are 5671 through 5676.*

- x. TS 4.5.2.2.a - This was revised to adjust the testing Frequency of the spray additive valves from monthly to once every 24 months consistent with SR 3.6.6.13. This increased testing interval is acceptable since the system only needs to be verified that it can actuate on an actual or simulated SI signal on a refueling basis similar to the SI and RHR systems. Any additional valve testing is addressed by the IST program. In addition, a new Surveillance (SR 3.6.6.9) was added to verify that the CS motor operated isolation valves actuate to their correct position once every 24 months following an actual or simulated SI signal. Finally, a new Surveillance (SR 3.6.6.14) was added to verify that the spray additive flow rate is within limits once every 5 years. These changes ensure that the CS and spray additive tank LCOs continue to be met. These are Ginna TS Category (v.b.38) changes.

[CTS32.x-L1]:

3.6Q162 New SR 3.6.6.13 is acceptable.

Status: [] Closed

Response: N/A

[CTS32.x-M1]:

3.6Q163 Adding a new SR 3.6.6.9 is acceptable.

Status: [] Closed

Response: N/A

[CTS32.x-M2]:

3.6Q164 Adding a new SR 3.6.6.14 is acceptable.

Status: [] Closed

Response: N/A

- xi. TS 4.5.2.3.3 and 4.5.2.3.4 - These were revised to require that each CRFC unit be operated for ≥ 15 minutes once every 31 days (SR 3.6.6.2). This test will ensure that the CRFC units are OPERABLE in accordance with the LCO. In addition, a new Surveillance is also required once every 24 months to ensure that the CRFC units start on an actual or simulated SI signal. These tests will ensure that the CRFC units are OPERABLE in accordance with the LCO. These are Ginna

TS Category (v.a) changes.

[CTS32.xi-M1]:

3.6Q165 Adding a new SR 3.6.6.2 is acceptable.

Status: [] Closed

Response: N/A

[CTS32.xi-M2]:

3.6Q166 Adding a new SR 3.6.6.11 is acceptable.

Status: [] Closed

Response: N/A

Section 3.7 Improved TS

76. ITS 3.7.1

- i. Table 3.7.1-1 was not added to the new specifications. The current Ginna Station accident analyses assume that all eight main steam safety valves (MSSVs) are available for pressure relief. No analyses have been performed at lower power levels to support the inoperability of one or more safety valves. Consequently, Table 3.7.1-1 and the second part of Condition B do not apply to Ginna Station. Required Action A.1 was also deleted and replaced with a requirement to restore an inoperable MSSV(s) to OPERABLE status within 4 hours consistent with current Ginna Station TS 3.4.1. The bases were revised to state that the 4 hour Completion Time is to address instances where the MSSVs are administratively declared inoperable since hardware related repairs cannot be performed during MODES 1, 2, or 3, similar to approved Traveller WOG-15, C.1 (Rev.1). These are ITS Category (i) changes.

[ITS76.i]:

3.7Q1

The proposed changes to this LCO are accepted; however, the above basis for explaining the changes to the BASES is not accepted and requires discussion. First, Traveler WOG-15, C.1 (rev. 1) does not exist but has been superseded by BWOG-09. Secondly, for Section 3.7 in NUREG-1431, neither of these travelers are specifically applicable here. This Ginna basis is presented as insert 3.7.1.2 in the proposed ITS BASES for SR 3.7.1.1. The above text should be deleted from here and this issue resolved under ITS76.iv.

Status: [] Open

Response: See response to 3.7Q8.

- ii. Table 3.7.1-2 was not added to the new specification since this table only provides the lift settings of the MSSVs. These values were relocated to SR 3.7.1.1 to consolidate the definition of MSSV OPERABILITY. That is, SR 3.7.1.1 now requires that the MSSVs have an "as left" lift setting within $\pm 1\%$ of the specified setpoint and an "as found" lift setting within $\pm 3\%$ of the specified setpoint. This is an ITS Category (i) change.

[ITS76.ii]:

3.7Q2

This change is acceptable; however, the tolerances above do not match the description provided in the accompanying BASES. The "as found" lift setting is $+1\%$ and -3% instead of $\pm 3\%$. This is acceptable but there should be text agreement. Also, why not add

the "as found" and "as left" terminology to the BASES to help clarify for the Ginna plant personnel?

Status: []

Open

Response:

The change justification is incorrect in that the OPERABILITY, "as found," acceptance limit is actually +1% and -3% as stated in the bases. Comment #92 has been opened to correct this error in Attachment A. Comment #93 has also been opened to clarify the bases as suggested.

- iii. The NOTE for SR 3.7.1.1 was revised to provide clarification that this Surveillance is only required to be performed prior to entry into MODE 2 from MODE 3 consistent with the NUREG-1431 bases. This is an ITS Category (iii) change.

[ITS76.iii]:

3.7Q3

This is acceptable; however; this change merely restates what the BASES already state is the basis for this note. Why make Ginna different from the standard?

Status: []

Open

Response:

The bases in the NUREG provide a different interpretation of SR 3.7.1.1 than the actual surveillance. As written in the NUREG Surveillance, the Note states that "Only required to be performed in MODES 1 and 2." Meanwhile, the bases state that this Note was added to allow "entry into and operation in MODE 3 prior to performing the SR." The Note in the surveillance could be interpreted as only being applicable in MODES 1 and 2, and never in MODES 3. This is not the intended interpretation since the SR is applicable in MODE 3 while de-powering. This issue is found throughout the NUREG and was identified by the Ginna Station Operations department. Comment #90 was opened during the review of Chapter 3.4 to address the use of "only required" in a SR note. Suggest this comment also track this issue. [See also comment #175]

- iv. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including providing consistency with current Ginna Station TS bases.
- b. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other basis sections as necessary.

[ITS76.iv.a and b]:

3.7Q4

In Background, the deletion of the second sentence in the second paragraph refers to the relief capacity of the MSSVs and not the design basis which has been reinserted. This is just a repeat of the first sentence in Applicable Safety Analyses. Suggest putting in what were the requirements of the applicable design code used at Ginna.

Status: []

Open

Response:

This level of information is not readily available, but will become available later next year upon complete of the ongoing design basis documentation (DBD) effort at Ginna. As such, RG&E recommends that the Background bases remain as proposed for the time being.

3.7Q5 In Applicable Safety Analyses, second paragraph - insertion of RCS does not seem important and change of AOO to DBA does not appear correct?

Status: []

Response:

Open

The "RCS" insertion provides clarification as requested during the initial internal review of these bases. The use of DBA in place of AOO is incorrect per the definition of AOO in 10 CFR 50, Appendix A. Comment #93 has been opened to revise the bases consistent with NUREG-1431 with respect to this issue.

3.7Q6 In LCO, deletion in the first paragraph, explain why the second and third sentences should not be restored except for "five" should be changed to "all" or "eight". Please explain in this case how the phrase "open on demand" is defined at Ginna. Is this an external signal to the valve to open, a passive pressure buildup that does not activate the valve lift, or something else? How is the Ginna MSSVs different from the Westinghouse standard design? Does Ginna have a power operated relief valve on each steam header with the MSSVs?

Status: []

Response:

Open

The Ginna MSSVs are the same as the Westinghouse standard design in that a spring design is used to open the relief valves. The second paragraph of the NUREG LCO bases state "the OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve SG overpressure, and reseal when pressure has been reduced." The deleted text states that "an MSSV will be considered inoperable if it fails to open on demand" and then proceeds to discuss reduced power requirements without the full complement of MSSVs. The "fails to open on demand" only duplicates text while the reduced power reduction issue was not added to LCO 3.7.1 as discussed above. The ARVs are located upstream of the MSSVs as shown on UFSAR Figure 6.2-76 and 6.2-77 (these were provided in the Chapter 6 review).

3.7Q7 In Applicability, insert 3.7.1.1 - explain why the phrase "to ensure that the RCS remains within its pressure safety limit" should not be deleted or if absolutely necessary, relocated at the end of the inserted text. The RCS has other more direct pressure controlling design features than just the MSSVs.

Status: []

Response:

Open

The inserted text at the top of page B 3.7-2 for the Applicable Safety Analyses states that at power levels > 50% RTP, the MSSVs and pressurizer safety valves are required to "maintain the RCS and Main Steam System within 110% of their design values." The Applicability reiterates this statement. Essentially, following a loss of external load event, if the MSSVs were unavailable, there is no place for the RCS to dump excess heat except through the pressurizer PORVs (which are not credited) and the safeties. The pressurizer safeties cannot remove the necessary heat such that the MSSVs are credited with their heat removal via the steam generators.

3.7Q8 In Actions, insert 3.7.1.2 is not accepted because it is implied that MSSVs never have inoperable hardware. They instead have non-significant discrepancies? or just have bad paperwork? The questions of item #3 above lead directly into the review of the contents of this insert. Also the four hour Completion Time to

include the NRC review is not practical for mere administrative problems. More explanation is required?

Status: []

Open

Response:

The 4 hour Completion Time is consistent with CTS 3.4.1. The CTS bases do not place any restriction on the use of this 4 hours (i.e., this 4 hours may be used for both hardware failures and non-significant discrepancies). The ITS bases state that the 4 hours may be used for non-significant discrepancies only and that hardware failures requires an immediate shutdown. This is a more restrictive change with respect to CTS requirements while WOG-15, C.1 was subsequently superseded by BWOG-09, this text is consistent with the original intent of that traveller for a similar issue.

- v. Incorporation of approved Traveller NRC-01, C.2. Verified as consistent with NUREG-1431, Rev. #1 BASES to ITS markup.

77. ITS 3.7.2

- i. The Applicability and bases were revised to require the MSIVs to be OPERABLE in MODES 1, 2, and 3 regardless of the position of the valve. The bases were also revised to state that an MSIV which is closed and de-activated is considered OPERABLE since the valve is in its assumed position for the accident analysis. As such, Conditions A and B are no longer applicable and were deleted. This change eliminates potential confusion and clarifies what is defined as an OPERABLE MSIV. This is an ITS Category (iii) change.

[ITS77.i]:

3.7Q9

Status: []

Response:

The change to applicable in MODES 1, 2, and 3 is acceptable.

Closed

N/A

3.7Q10

The change stated in the second sentence is not acceptable here nor in ITS LCO 3.6.3. This negates the definition in Section 1.0. This is merely the taking of alternate action to maintain the assumptions of the accident analyses when a disabled system cannot function on its own.

Status: []

Open

Response:

The NUREG is very confusing with respect to the MSIV MODE of Applicability. The NUREG Condition D requires entry into MODE 4 within 12 hours if the previous Required Actions are not met. However, the actual MODE of Applicability is "MODES 1 and MODES 2 and 3 except when all MSIVs are closed and [deactivated]." If in MODE 2 or 3, the plant should be provided with the option to close and deactivate the MSIV similar to every other LCO when the Required Actions are not met. In addition, while in MODES 2 and 3 with the MSIV closed and deactivated, the MSIVs are in fact OPERABLE and performing their safety function. With a MODE of Applicability stating that if a MSIV is closed and deactivated it is not required to be OPERABLE is misleading. [This response was revised during meetings the week of 10/9/95. See comment #134.]

3.7Q11

It is acceptable to delete Condition B; however, Condition A stays. Condition A is rewritten to state "One MSIV and/or one non-return check valve inoperable on the same steam header in MODE 1". The

Required Action is "Restore the valve(s) to OPERABLE status" with a Completion Time of 8 hours. Proposed new B is rejected because multiple condition entry would allow proposed Condition A and B to be entered simultaneously to result in Condition D. Therefore, Condition D is not needed; since by omitting it, this is an LCO 3.0.3 situation. Please note the resolutions proposed to rewritten Condition C in ITS77.ii.

Status: []

Response:

Open

The ITS Writer's Guide does not allow the use of "and/or" in a Condition or Required Action statement. Therefore, the proposed resolution cannot be implemented. In addition, ITS Conditions A and B only result in Condition D if they occur on opposite steam lines. If Conditions A and B exist on the same steam line, Condition D does not apply since there is no loss of safety function as the remaining MSIV can close and isolate the steam generators from each other. The use of multiple condition entry in this instance is no different from any other LCO. For example, for a two train system, there is typically a condition specified if you lose one train. However, if you lose both trains, you are in LCO 3.0.3 plus the condition which applies to the loss of one train.

- ii. The Completion Time for Required Action C.1 was changed from 8 hours to 24 hours. The current Ginna Station TS do not contain any Required Actions with respect to an inoperable MSIV. A Completion Time of 24 hours was selected to allow restoration of an inoperable MSIV due to the ability to isolate a SG by other means (e.g., turbine stop valves). This is an ITS Category (i) change.

[ITS77.ii]:

3.7Q12

Condition C statement is counter proposed as Condition B to be "One or more MSIVs and/or non-return check valves inoperable in MODES 2 or 3" with the Required Actions and Completion Times remaining the same to be consistent with Condition A.

Status: []

Response:

Open

The ITS Writer's Guide does not allow the use of "and/or" in a Condition or Required Action statement. Therefore, the proposed resolution cannot be implemented.

3.7Q13

At Ginna, how is a SG isolated by the turbine stop valves without isolating both SGs?

Status: []

Response:

Open

The turbine stop valves can isolate a steam line break (SLB) in the Turbine Building located downstream of these valves such that the MSIVs are not required. In addition, if a MSIV were to fail, its associated non-return check valve would prevent a blowdown of the unaffected steam generator if a SLB were to occur inside containment.

- iii. The Note for Condition C was not added to the new specifications since Ginna Station only has two installed MSIVs. Consequently, if both MSIVs are inoperable, the plant is outside the accident analysis in the event of a SLB. This is also true if both non-return check valves are inoperable, or one or more isolation valves from each SG are inoperable. The description for Condition C was also revised to limit its application to only one inoperable MSIV.

A new Condition was added in the event that one or more isolation valves from each SG are declared inoperable requiring entry into LCO 3.0.3. These are ITS Category (iv) and (iii) changes respectively.

[ITS77.iii]:

3.7Q14 This note can now be added as originally in NUREG-1431.

Status: [] Open

Response: *RG&E disagrees. As stated in the change justification, if both MSIVs are inoperable in MODES 1, 2, or 3, the plant is outside the accident analyses with respect to a SLB. This Note was added with respect to plants with 4 SGs and 4 MSIVs. With only 2 SGs and 2 MSIVs, this Note is not applicable. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

3.7Q15 Since this rewrite of Condition C applies in only MODES 2 or 3, two MSIVs can be inoperable and closed to maintain the accident analysis assumptions.

Status: [] Open

Response: *See response to 3.7Q12. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

3.7Q16 One inoperable MSIV is now under Condition A.

Status: [] Open

Response: *One MSIV inoperable is addressed both in the ITS Condition A and the reviewer proposed Condition A. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

3.7Q17 As noted in ITS77.ii, new Condition D is not necessary. Old Condition D becomes C just as proposed in this submittal.

Status: [] Open

Response: *RG&E disagrees. Please see response to 3.7Q11. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

- iv. The Completion Time for Required Action C.2 was revised from once every 7 days to once every 31 days. The current Ginna Station TS do not contain this requirement. A Frequency of once every 31 days is consistent with the Required Actions for LCO 3.6.3 which is appropriate since the MSIVs also perform a containment isolation barrier function as described in the bases. This is an ITS Category (i) change.

[ITS77.iv]:

3.7Q18 This is rejected because the length of time to reverify is insignificant to the real effort which should be to restore the MSIVs or non-return check valves OPERABLE and to return to MODE 1.

Status: [] Rejected

Response: *There is a WOG generated traveller on this issue scheduled to go to the NRC by November 1st. Comment #108 has been opened to track this issue. Suggest change status to "Open" until this traveller is resolved. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

- v. SR 3.7.2.1 was revised to require that each MSIV be tested under no

flow and no load conditions consistent with current Ginna Station TS 4.7. This is a conservative test since the valve is assisted in closing when steam flow is available. As such, the valve closure time under hot conditions would be dependent upon available steam flow. In addition, a new Surveillance (SR 3.7.2.2) was added which requires verification once every 24 months that each MSIV can close on an actuation signal, independent of closure time, consistent with the accident analysis assumptions and current testing practices. These are ITS Category (i) changes.

[ITS77.v]:

3.7Q19

Status: [] Closed

Response: N/A

SR 3.7.2.1 is acceptable.

3.7Q20

Status: [] Open

Response:

SR 3.7.2.2 is not new to Ginna.

Correct, SR 3.7.2.2. is not new to Ginna Station but is new with respect to the NUREG which is where this change is discussed.

3.7Q21

Status: [] Open

Response:

The questions to ITS77.viii need answers to separate the non-return check valves from the MSIV function.

See responses to 3.7.33 through 3.7.36.

3.7Q22

Status: [] Open

Response:

Since the non-return check valves are not in the existing TS, could testing under no flow and no load for MSIV be mainly for verifying the non-return check valve? What are the test conditions for new SR 3.7.2.3? Is this conservative or non-conservative?

A copy of the test procedure for ITS SR 3.7.2.3 is attached (PT-2.10.15). Essentially, this procedure verifies that the non-return check valves are closed if the associated MSIV has closed. This test is performed during any condition, hot or cold, with the associated MSIV closed. Testing under no flow and no load conditions prevents unnecessary slamming of the MSIV and is conservative since flow in the main steam lines assists the MSIV in closing by design. Therefore, if the MSIV isolation times can be met under no flow conditions, the test is bounding with respect to accident analysis assumptions. Testing under no flow and no load conditions for the non-return check valves is not specified although it is allowed by this new SR. However, this is also conservative since steam flow from the opposite steam generator will assist in closure of the check valve.

vi. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including providing consistency with current Ginna Station TS bases and the accident analyses. As such, approved Traveller WOG-24, C.3 was not added.

b. Various wording changes were made to improve the readability and understanding of the bases.

- c. Discussion was added related the function of the MSIVs with respect to containment isolation.

[ITS77.vi.a, b and c]:

3.7Q23 What is in WOG-24, C.3 that is not added?

Status: [] Open

Response: *Traveller WOG-24, C.3 deleted selective text in item d. of the Applicable Safety Analyses bases. Since the entire item d. is proposed to be deleted, this Traveller was not incorporated at all. [This response was revised during meetings the week of 10/9/95. See comment #134.]*

3.7Q24 As noted in ITS77.viii, the function of the non-return check valve is not adequately described to incorporate into the LCO as yet. As changes are pending, until this is resolved. In ZYINDEX, a non-return check valve does not exist in the UFSAR?

Status: [] Open

Response: *As seen in the procedure provided in response to 3.7Q22, Ginna Station identifies these as non-return check valves. These non-return check valves are discussed in the second to last paragraph of UFSAR Section 15.1.5.1.2.*

3.7Q25 Please verify the MSIV bypass valve is not a check and it is manually opened/closed only. Is it a containment isolation valve? Does the MSIV bypass valve loop include the MSIV and the non-return check valve?

Status: [] Open

Response: *The MSIV bypass valve is a normally closed manual valve as stated in the last sentence for the Background bases. This valve is identified as a containment isolation valve in UFSAR Table 6.2-14 and as shown on UFSAR Figures 6.2-76 and 6.2-77 (previously provided in response to Chapter 3.6 questions).*

3.7Q26 "The MSIVs may also be actuated manually." is added twice.

Status: [] Open

Response: *"The MSIVs may also be actuated manually" should be deleted from the ballooned text on the left of the page. Comment #94 has been opened to correct this error.*

3.7Q27 Should insert 3.7.2.2 be justified as 77.vi.a?

Status: [] Open

Response: *The justification "77.iv.a" is a typographical error as noted. Comment #94 has been opened to change this justification to "77.vi.a."*

3.7Q28 Shouldn't insert 3.7.2.3 read ", high steam flow and 2 out of 4 low T_{avg} coincident with safety injection (SI), or high-high steam flow coincident with SI." per existing TS Table 3.5-2, Functional Unit 5.a and b?

Status: [] Open

Response: *Insert 3.7.2.3 is consistent with ITS Table 3.3.2-1, Function #4.d. Therefore, no change is necessary.*

3.7Q29 Insert 3.7.2.4 has a different accident assumptions than the

standard MSIV BASES in second paragraph. In the SLB for containment integrity analysis, offsite power is not assumed to be available but it is available for Ginna. In the third paragraph of the SLB inside of the turbine building, both MSIVs are assumed to isolate but what about a single failure of one? Please explain these differences and how it could affect the LCO?

Status: []

Open

Response:

For the NUREG containment analysis, loss of offsite power is assumed with failure of the associated MSIV to close such that reverse flow results from the main steam header into containment. However, with use of the non-return check valves, reverse flow is not assumed. Consequently, blowdown from only the affected steam generator is postulated. With offsite power available, the reactor coolant pumps assist in providing forced RCS flow through the affected steam generator; hence more steam from the steam generator and a higher containment pressure. The loss of offsite power causes the loss of forced RCS flow and less steam generation.

With respect to a SLB in the turbine building, if either MSIV fails to close, the associated steam generator continues to feed the break until the feedwater source to the steam generator is isolated. This is no different from a SLB upstream of the MSIVs. The bases statement is intended to show that the MSIVs, and not the non-return check valves, are credited in this instance.

3.7Q30

In LCO, as noted in ITS77.i, inoperable MSIVs that are closed and de-activated are not consider OPERABLE.

Status: []

Open

Response:

See response to 3.7Q10.

3.7Q31

Comments to BASES in Actions and Surveillance Requirements are dependent on resolutions reached in the LCO.

Status: []

Open

Response:

See responses to 3.7Q9 through 3.7Q30.

vii. Incorporation of approved Traveller NRC-01, C.2.

[ITS77.vii]:

3.7Q32

This 77.vii was not marked in ITS. Does it apply to Reference #5 of BASES?

Status: []

Open

Response:

Correct, "77.vii" applies to Bases Reference 5. Comment #94 has been opened to correct this.

viii. The LCO was revised to add requirements and surveillances for non-return check valves which are in-series with each MSIV. These non-return check valves are credited in the accident analysis are therefore added to the new specifications. The title and bases were also appropriately revised. These are ITS Category (ii) changes.

[ITS77.viii]:

3.7Q33

From the BASES changes, it appears that without the non-return check valves, the MSIVs could not function as more recent MSIVs designs which can isolate to prevent back flow of steam in the main steam header from emptying into containment or isolating the unaffected SG. It appears the non-return check valve is part of the MSIV

function and should they be treated as one?

Status: []

Open

Response:

The Ginna MSIVs can function like all other Westinghouse MSIVs. However, most Westinghouse plants do not have an installed non-return check valve. Since Ginna was designed with this check valve, the accident analyses have credited their function. As stated in the Applicable Safety Analyses bases, the non-return check valves are only credited in the SLB with respect to containment integrity. For all other accident scenarios, the MSIVs are credited in the accident analyses. See UFSAR Figures 6.2-76 and 6.2-77 for the configuration of the MSIVs and non-return check valves.

3.7Q34

How are these valves currently treated at Ginna?

Status: []

Open

Response:

See response to 3.7Q33. In addition, the non-return check valves are tested each cold shutdown prior to plant startup per procedure PT-2.10.15 (attached).

3.7Q35

Are the MSIVs and non-return check valves separately tested or together?

Status: []

Open

Response:

See response to 3.7Q22.

3.7Q36

Is the non-return check valve designed to the same requirements as the MSIV? Is it designated as a containment isolation valve for this penetration?

Status: []

Open

Response:

See response to 3.7Q33. Since the non-return check valves are located downstream of the MSIVs, they are not identified as containment isolation valves.

- ix. Various editorial changes were made which provide clarity but do not alter the intent of the LCO. These are ITS Category (iv) changes.

[ITS77.ix]:

3.7Q37

This was only noted once for Required Action C.1 which was rejected in ITS77.ii.

Status: []

Open

Response:

See response to 3.7Q12.

3.7Q38

Please provide a list where the other various changes were made so they can be evaluated.

Status: []

Open

Response:

Change 77.ix was only used to add "inoperable" to NUREG Required Action C.1. Comment #92 has been opened to revise this change justification to read "Required Action C.1 was revised to provide clarity but not alter the intent of the LCO. This is an ITS Category (iv) change."

78. ITS 3.7.3

- i. The title was revised to be consistent with Ginna Station nomenclature which includes the use of "main feedwater pump discharge valve (MFPDV)." This is an ITS Category (iv) change.

[ITS78.i]:
3.7Q39

In MFPDV, the removal of the word "isolation" from this valve name is important to the purpose for this LCO. There appears to be no unique use of the proposed new valve name after reviewing UFSAR 10.4.5, Feedwater System. MFIV from the NUREG-1431 would clearly apply to the motor-operated isolation valve immediately downstream of the main feedwater pump. The check valve between the pump and the isolation valve would be the "main feedwater pump discharge valve". There are no pump numbers identified. Also, there appears to be no associated bypass valve with the isolation valve. There appears to be only one bypass valve associated with each MFRV. This needs correcting in the LCO title, LCO and BASES text.

Status: []
Response:

Rejected
The use of MFPDV is consistent with Ginna Operations procedures who requested the use of this title (see attached procedure 0-1.2, pages 62 and 66). Since operators are the actual end user of the TS, RG&E requests the use of this title. The check valve between the pump and isolation valve is the "main feedwater pump discharge check valve." See also the attached sketch. With respect to pump numbers, the NUREG does not identify pump numbers in the LCO or bases; hence RG&E has not provided this identification. However, since there are only two MFW pumps at Ginna, this is not a concern. The reviewer is correct in that there are no bypass valves associated with the MFPDV, only with respect to the MFRV. RG&E proposes to revise title to read "MFRV and Associated Bypass Valves and MFPDVs." Comment #95 has been opened to address this.

- ii. Condition D was not added since the current Ginna Station TS do not contain these requirements. The fact that two parallel valves are inoperable should not require a shorter isolation time since containment isolation penetrations do not have similar requirements. This is an ITS Category (i) change.

[ITS78.ii]:
3.7Q40

Condition D is written to override multiple condition entry into Condition A, B and C at the same time that results in a main feedwater flowpath being unable to isolate, if needed to meet the accident analyses assumptions. If this is deleted then extensive rewrite is required. The shorter time is justified because should the containment be breached, there is no way to automatically isolate this flowpath.

Status: []
Response:

Rejected
Proposed Condition E prevents being outside the accident analysis assumptions. In addition, The MFPDV and MFRV are not containment isolation valves at Ginna Station. Instead, a check valve located between the MFRV and containment is the containment isolation valve for these penetrations (see attached sketch). In addition, if the MFPDV and MFRV failed to close, the MFW pumps can be manually tripped to provide feedwater isolation in combination with the two containment isolation check valves.

- iii. The Completion Time for Required Actions A.2, B.2, and C.2 was revised from once every 7 days to once every 31 days. The current Ginna Station TS do not contain this requirement. A Frequency of once every 31 days is considered acceptable due to the available

indications of valve position available to plant operators. This is an ITS Category (i) change.

[ITS78.iii]:

3.7Q41 Though not mentioned above, the 24-hour Completion Time for Required Action A.1, B.1, and A.3 is accepted.

Status: [] Closed

Response: N/A

3.7Q42 Though not mentioned above, the text changes to Required Actions A.1, A.2, C.1, and C.2 are rejected because "close" is not same as isolated.

Status: [] Open

Response: See response to comment 3.7Q51.

3.7Q43 The change in Completion Time from 7 days to 31 days is not accepted because this is the main feedwater system which is meant to be OPERABLE and open rather than isolated and relying on the safety backup auxiliary feedwater system for normal operation. Also the 31 day frequency is a visual verification as is done for LCO 3.6.3.

Status: [] Open

Response: *There is a WOG generated traveller on this issue scheduled to go to the NRC by November 1st. Comment #108 has been opened to track this issue. With respect to the comment concerning use of MFW and not relying on AFW, the MODE of Applicability for LCO 3.7.3 is MODES 1, 2, and 3. MFW is actually only in service during MODE 1, above 5% RTP. At all times below this power level, AFW is the normal source of feedwater.*

iv. A new Condition was added in the event that both MFW flowpaths to the SGs have at least one inoperable valve. The new Condition requires entry into LCO 3.0.3 since the plant is outside the accident analyses. As a result of this addition, Condition E was revised to specify that it would only be entered in the event that the Required Actions of Condition A, B, or C were not satisfied consistent with the ITS Writer's Guide. This is an ITS Category (iii) change.

[ITS78.iv]:

3.7Q44 New proposed Condition E is not understood as justified above. The first sentence says a new condition was added and then the third sentence says it was subsequently revised? The above justification implies that Ginna must have at least one operable flowpath with water flowing to the steam generator. The BASES to this condition read that there must not exist a flowpath which is unisolable to one or both SGs. The deleted Condition D was written for this later implied purpose for new Condition E. ????

Status: [] Open

Response: *Due to the previous changes to the NUREG, Condition E of the NUREG was revised to be Condition D. As such, when the new Condition was added, it became the new Condition E. The above justification states that if the two MFW paths to the steam generators (i.e., one path per steam generator) each have one or more inoperable valves, then LCO 3.0.3 must be entered. This is different from Condition D which requires an early shutdown path if two valves to the same*

steam generator are inoperable. A sketch of the MFW isolation valves is attached. From this figure it can be seen that there are not really two isolation valves per steam generator such that Condition D is not applicable to Ginna as written in the NUREG.

- v. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Plant-specific design considerations were added including providing consistency with the accident analyses.
 - b. Various wording changes were made to improve the readability and understanding of the bases. This includes the deletion of text that is not related to the technical specification function performed by the MFRVs and bypass control valves.

[ITS78.v.a and b]:

3.7Q45 As noted in ITS78.i above, the name changes are not sufficiently justified. The term "associated" apparently only applies to the MFRV but the BASES title and text imply it also includes the MFIV. This needs modification throughout background.

Status: [] Open

Response: See response to 3.7Q39.

3.7Q46 In LCO, the deletion of the last sentence of the first paragraph should be explained. Isn't the main feedwater line safety related from the SG to the MFRV? Where is the boundary?

Status: [] Open

Response: At Ginna Station, check valves 3992 and 3993 provide the safety class boundary since these are the containment isolation valves for the MFW penetrations (see UFSAR Figure 6.4-78).

3.7Q47 In LCO, the text addition at the end of the second paragraph is rejected based upon ITS78.vi below.

Status: [] Open

Response: See response to 3.7Q51.

3.7Q48 In LCO, explain deletion in third paragraph.

Status: [] Open

Response: As explained in the ITS Background bases, the MFPDVs only close on opening of the MFW pump breakers which are opened on a SI signal or low pump suction pressure. Meanwhile, the MFRVs close on a SI signal, high steam generator level, or on reactor trip with $T_{avg} < 554^{\circ}\text{F}$ with the valve in auto. The accident analyses only credit the SI signal with respect to MFW isolation and not the high steam generator level as stated in the NUREG bases. Also, since the SI signal does not directly close the two MFPDVs, it was decided not to revise the bases text to replace "high steam generator level" with "SI signal." Therefore, the text was deleted.

3.7Q49 Comments to the BASES Actions are deferred until the contents of the LCO Condition statements are resolved.

Status: [] Open

Response: See response to 3.7Q39 through 3.7Q48.

- vi. The Applicability and bases were revised to require the MFRVs and bypass valves to be OPERABLE in MODES 1, 2, and 3 regardless of the position of the valves. The bases were revised to state that a valve which is closed and de-activated, or isolated by a closed manual valve, is considered OPERABLE since the valve is in its assumed position for the accident analysis. This change eliminates potential confusion and clarifies what is defined as an OPERABLE MFRV and bypass valve. This is an ITS Category (iii) change.

[ITS78.vi]:

3.7Q50 It is acceptable to require these OPERABLE in MODES 1, 2, and 3.

Status: [] Closed

Response: N/A

- 3.7Q51 The change stated in the second sentence is not acceptable here, nor in ITS LCO 3.6.3 or ITS77.i above. This negates the definition in Section 1.0. This is merely the taking of alternate action to maintain the assumptions of the accident analyses when a disabled system cannot function on its own. This change adds confusion rather than clarifies. The BASES must be returned to the original text.

Status: [] Open

Response: *The NUREG is very confusing with respect to the MFW isolation valve MODE of Applicability. Condition E requires entry into MODE 4 within 12 hours if the Required Actions are not met. However, the actual MODE of Applicability is "MODES 1 and MODES 2 and 3 except when MFIV, MFRV is closed and [de-activated][or isolated by a closed manual valve]." The plant should be provided with the option to close and deactivate the MSIV similar to every other LCO in which the previous Required Actions are not met. In addition, with these isolation valves closed and deactivated, the valves are in fact OPERABLE and performing their safety function. With a MODE of Applicability stating that if an isolation valve is closed and deactivated it is not required to be OPERABLE is misleading.*

- vii. SR 3.7.3.1 was separated into two surveillances since the MFPDVs have a different isolation time (as assumed in the accident analysis) than the other isolation valves. This is an ITS Category (iv) change. Acceptable

.79. ITS 3.7.4

- i. The title was revised to be consistent with Ginna Station nomenclature which includes the use of "atmospheric relief valve (ARV)" versus "atmospheric dump valve (ADV)." This is an ITS Category (iv) change. Acceptable

- ii. The LCO, Conditions, Required Actions, Surveillances and bases were revised since the ARVs at Ginna Station do not have a remote operated block valve. The spurious opening of an ARV is considered within the accident analyses such that a block valve is not required. As such, SR 3.7.4.2 was not added. This is an ITS Category (iv) change.

[ITS79.ii]:

3.7Q52 The LCO is to verify the OPERABLE status of the atmospheric relief

valve flowpath under both containment isolation conditions and alternate use as a steam dump path. Therefore, all components in this flowpath must be operable to meet the OPERABLE status for this LCO. The NUREG-1431 does not require the block valve to be remotely operated. A manual valve is acceptable. The assumed time to close this valve is a factor in the accident analyses assumptions. Therefore, please add in the "line" to ARV removed from the descriptions for this LCO.

Status:[]

Open

Response:

RG&E agrees to add "line" back into the OPERABILITY requirements for the ARV since the manual valve is required to isolate a failed ARV when it is being used during a SGTR. Comment #93 has been opened to address this.

3.7Q53

SR 3.7.4.2 should be performed as required.

Status:[]

Open

Response:

RG&E agrees to add this SR to the ITS. Comment #96 has been opened to address this.

- iii. The Applicability and Required Actions C.1 and C.2 were revised to only require the ARVs in MODES 1 and 2 and when the RCS average temperature is > 500°F in MODE 3. At Ginna Station, the ARVs are only credited in the accident analyses with respect to providing cool down capability following a SGTR in order to maintain subcooling margin. With the RCS average temperature < 500°F, the saturation pressure of the primary system is below the MSSV setpoints and the ARVs are not required. See the new bases for additional information. This is an ITS Category (i) change.

[ITS79.iii]:

3.7Q54

This appears acceptable. Please verify that the additional reason this is different from the NUREG-1431 is also that the ARVs are not used for cool down to the RHR crossover temperature at 350°F.

Status:[]

Open

Response:

Cooldown to the RHR entry temperature does require some form of steam relief from the steam generators. This can be accomplished via the ARVs or the steam dump system. However, this cooldown is not addressed in any accident analyses since Ginna was designed and analyzed to show that hot shutdown could be achieved and maintained. Additional cooldown to RHR entry conditions was, and is not required.

- iv. Condition B was revised to require entry into LCO 3.0.3 immediately when both ARVs are inoperable. Since Ginna Station only has two ARVs, the inoperability of both valves would result in the loss of a safety function as assumed in the accident analyses. This is an ITS Category (iv) change.

[ITS79.iv]:

3.7Q55

The Required Action of rewritten Condition B (now C) could be "Be in MODE 3 in 6 hours" and in "Be in MODE 5 in 30 hours". This the more direct way of entering an LCO 3.0.3 shutdown. It is the OTSB general policy not to prescribe entering LCO 3.0.3.

Status:[]

Open

Response: Requiring entry into LCO 3.0.3 is used elsewhere in the NUREG (see LCOs 3.3.1 and 3.5.1). Also, the addition of this LCO 3.0.3 entry throughout the ITS was discussed during the review of Chapters 3.1, 3.2, 3.4, and 3.5 and found to be acceptable to the NRC. RG&E operators prefer the direct mention of LCO 3.0.3 since it is clear that there is a loss of safety function. Also, 1 hour is allowed in LCO 3.0.3 to prepare for a plant shutdown which would not be the case in the proposed rewrite of the Required Actions.

- v. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Plant-specific design considerations were added including providing consistency with the accident analyses.
 - b. Various wording changes were made to improve the readability and understanding of the bases. This includes the deletion of text that is not related to the technical specification function performed by the ARVs.

[ITS79.v.a and b]:

3.7Q56 In insert 3.7.4.1, the ARVs are located in the intermediate building on a tapped line off of the steam header downstream of the MSSVs and are not "on each SG".

Status: [] Open

Response: RG&E agrees to revise the first sentence of Insert 3.7.4.1 to read "There is an ARV (3410 and 3411) located on the main steam header from each steam generator (SG)." Comment #93 has been opened to address this.

3.7Q57 In Background, the third paragraph deleted should be added after the insert 3.7.4.1.

Status: [] Open

Response: RG&E agrees to retrain the third Background bases paragraph from the NUREG. Comment #93 has been opened to address this.

3.7Q58 In Applicable Safety Analyses, explain deletion of second paragraph. Add in deleted third paragraph.

Status: [] Open

Response: Cooldown to the RHR entry temperature does require some form of steam relief from the steam generators. This can be accomplished via the ARVs or the steam dump system. However, this cooldown is not addressed in any accident analyses since Ginna was designed and analyzed to show that hot shutdown could be achieved and maintained. Additional cooldown to RHR entry conditions was, and is not required. With respect to the deleted third bases paragraph, RG&E agrees to add this text with the addition of "following a SGTR event" at the end of the sentence. Comment #93 has been opened to address this.

3.7Q59 In LCO, first paragraph the third and fourth sentences deleted are applicable here. Also explain insert 3.7.4.5 in lieu of deleted text.

Status: [] Open

Response: RG&E agrees to add the deleted third and fourth sentences of the

first LCO bases paragraph. Comment #93 has been opened to address this. The Insert 3.7.4.5 text is the actual times with respect to the use of the ARVs following a SGTR event as taken from operator training material. This text states that the ARVs must be capable of both opening and closing within the prescribed times to be considered OPERABLE.

3.7Q60 In Actions A.1, the added text at the end seems misplaced and should be at the end of the previous sentence. Why are MSSVs deleted here?

Status: [] Open

Response: *The added text at the end of Required Action A.1 is to explain why the Note with respect to LCO 3.0.4 has been added. The NUREG only provides a statement that the Note exists, not why the Note was added. The change was requested by Ginna operators. The reference to MSSVs was deleted since these valves have setpoints above the ARVs which are not adjustable during power operation or accident conditions such that they cannot be credited with respect to a SGTR event. The fact that the MSSVs are OPERABLE has no impact with respect to the ARV OPERABILITY requirements. As such, RG&E believes the proposed markup is correct.*

3.7Q61 In Completion Time for new B.1 from 6 hours to 8 hours has not been separately explained as why longer time is appropriate.

Status: [] Open

Response: *The longer time is appropriate for ITS Required Action B.1 since the required MODE to be entered is "MODE 3 with $T_{avg} < 500^{\circ}F$ " versus only "MODE 3." The 8 hours to perform this action is consistent with ITS LCO 3.4.16. Comment #97 has been opened to revise Change 79.iii to state this.*

vi. The text of SR 3.7.4.1 was revised to provide consistency with other similar tests (see SR 3.4.11.1) and the bases. This is an ITS Category (iii) change. Acceptable

80. ITS 3.7.5

i. The Auxiliary Feedwater (AFW) System at Ginna Station is comprised of two systems, a preferred AFW System and a Standby AFW (SAFW) System. Each system provides a portion of the overall AFW System function. The LCO, Conditions, Required Actions, Surveillances, and bases were all revised to reflect the functions of the preferred AFW and SAFW Systems as described in the new bases. The Conditions, Required Actions, and their Completion Times were also significantly revised consistent with current Ginna Station TS 3.4.2. However, several changes from the current Ginna Station TS were also made to provide consistency with the accident analyses and for human factor reasons. These changes are discussed in detail in Section D, item 14.ii. New Surveillances were also added with respect to the SAFW System consistent with current Ginna Station TS 4.8. Reference 28 provides additional information. These are Category (i) and (ii) changes.

[ITS80.i]:
3.7Q62

NUREG-1431 was developed with multiple condition entry. The

proposed ITS does not work as written because new Conditions D and E are the same as new Condition G with different Required Actions. A rewrite is required to either achieve multiple condition entry, create single condition entry conditions or make two LCOs. The balance of comments on this LCO are dependent upon the format selected.

Status: []

Open

Response:

RG&E agrees that the proposed LCO is broke. A simple solution would be to add a Note to Conditions D and E saying they are not applicable if both AFW and SAFW are inoperable. However, suggest that this LCO be discussed in detail during the planned Ginna site visit. [Resolved per comment #139]

3.7Q63

Please confirm that Ginna does not want a condition for an inoperable steam flowpath to the TDAFW pump per old Condition A.

Status: []

Open

Response:

An inoperable steam flowpath to the TDAFW pump is contained in ITS Condition A. The bases for Required Action A.1 states that a "turbine driven AFW train flowpath is defined as the steam supply line and the SG injection line from/to the same SG" (Insert 3.7.5.9). This bases definition was added since Ginna Station has one TDAFW pump which feeds both SGs and receives steam from both SGs. As such, the inoperability of a steam supply from a SG is no different from the inoperability of the TDAFW supply to the same SG. ITS Condition A addresses both of these flowpaths. [Resolved per comment #139]

3.7Q64

The Conditions need to be arranged so the Completion Times are in descending order.

Status: []

Open

Response:

The Conditions were ordered mainly on the NUREG order which does not sort the Conditions based on Completion Times. Suggest discussing this issue during the planned Ginna site visit.

- ii. The LCO Note and Condition E were not added, and the Applicability and associated bases were revised to only apply in MODES 1, 2, and 3 consistent with current Ginna Station TS 3.4.2. The requirement for AFW during MODE 4 when the SGs are being relied upon for heat removal is controlled by new LCO 3.4.6 which specifies required SG level requirements. Due to the wide variety of means of providing decay heat removal in MODE 4 (e.g., AFW, SAFW, MFW, condensate booster pumps), RG&E does not believe that it is necessary to specify AFW requirements. This is also discussed in Reference 26. This is an ITS Category (i) change. As such approved Traveller WOG-27, C.1 was only partially incorporated (see also Section C, item 80.iv below).

[ITS80.ii]:

3.7Q65

UFSAR 10.5.3.1.2 clearly states that AFW is used to maintain steam generator water level during MODE 4 operations as the reactor is being shutdown. Therefore, this AFW Applicability requirement and the other LCO deleted requirements must be restored and/or rewritten to add provisions for a SAFW train.

Status: []

Open

Response:

There is a WOG Traveller on this issue which is scheduled to be

submitted to the NRC on November 1st. Comment #99 has been opened to track this Traveller. While the AFW System is the preferred means of supplying feedwater to the steam generators as stated in the UFSAR, there are numerous other sources as discussed in the justification above. The proposed MODE of Applicability is also consistent with CTS.

- iii. The Note for SR 3.7.5.2 and SR 3.7.5.3 was revised to require the turbine driven AFW pump to be tested prior to entering MODE 1 consistent with current Ginna Station TS 4.8.6. This is also discussed in Reference 28. This is an ITS Category (i) change.

[ITS80.iii]:

3.7Q66 The revised note per insert 3.7.5.2 is acceptable for SR 3.7.5.2 and the new SR 3.7.5.6 only and not for proposed SR 3.7.5.3.

Status: [] Open

Response: *This is a typographical error in Attachment A to the submittal in that the Note for NUREG SR 3.7.5.2 and SR 3.7.5.4 were revised, not SR 3.7.5.3. Comment #92 has been opened to correct this error.*

3.7Q67 The existing TS relaxation to the SR 3.7.5.2 interval is not justified and it is rejected.

Status: [] Rejected

Response: *There is a WOG Traveller on this issue which is scheduled to be submitted to the NRC on November 1st. Comment #98 has been opened to address the missing justification in Attachment A and to track this traveller.*

3.7Q68 The new proposed SRs 3.7.5.3 and 3.7.5.4 are acceptable except the interval shall be same as SR 3.7.5.2.

Status: [] Open

Response: *There is a WOG Traveller on this issue which is scheduled to be submitted to the NRC on November 1st. Comment #98 has been opened to address the missing justification in Attachment A and to track this traveller.*

- iv. SR 3.7.5.3 and SR 3.7.5.4 were revised to delete "when in MODE 1, 2, or 3" from the end of the Surveillance description consistent with approved Traveller WOG-27, C.1. The bases state that these Surveillances should only be performed during shutdown conditions to prevent the possibility of creating a plant transient. The deleted text implies that this test should be only performed in MODE 1, 2, or 3. The intention of the two SRs is to ensure that AFW will correctly actuate when it is in its MODE 1, 2, or 3 configuration. Specifying this in the Surveillance is inconsistent with all other SRs (e.g., ECCS) and is unnecessary. This is an ITS Category (iii) change.

[ITS80.iv]:

3.7Q69 It is acceptable to remove the text implying this test is only in MODES 1, 2 or 3 per the traveler.

Status: [] Closed

Response: NA

3.7Q70 A new note shall be added to new SRs 3.7.5.5 and 3.7.5.6 because AFW

is used during MODE 4. The note is "This SR is not applicable in MODE 4 when steam generator is relied upon for heat removal." This is from NUREG-1431 Rev.1.

Status: [] Open

Response: See response to 3.7Q65.

3.7Q71

Does this justification suggest there are other new SR 3.7.5.5 text changes similar to new SR 3.7.5.6 which are needed and are not marked in the BASES? The text assumed an ESFAS signal would initiate this test rather than a simulated actuation signal. How does Ginna plan to conduct this test?

Status: [] Open

Response: *The AFW pumps at Ginna are actuated from an ESFAS signal, low steam generator level, undervoltage conditions on the RCP buses, and opening of the MFW pump breakers (see ITS Table 3.3.2-1). Currently, all of these test signals are verified once each refueling outage (see attached procedure RSSP-3.0). The deleted bases text related to "ESFAS signal" for ITS SR 3.7.5.6 ensures that there is no misinterpretation with respect to what signals are to be verified (i.e., all of the applicable start signals are now required to be verified). The only AFW automatic valves are the AFW pump discharge valves (MOVs 4007 and 4008 for the motor driven pumps and AOVs 4297 and 4298 for the TDAFW pump). These valves do not receive an open signal from ESFAS, low steam generator level, etc. Instead, these valves open upon closing of the AFW pump breakers. Therefore, the SR 3.7.5.5 bases text related to "ESFAS signal" is not applicable.*

v. SR 3.7.5.5 was not added to the new specifications. The current Ginna Station TS do not contain this requirement. The verification of the correct lineup of the AFW and SAFW Systems is performed by SR 3.7.5.1. In addition, the AFW System takes suction from the CSTs during normal startup and shutdown conditions. The bases for SR 3.7.5.5 states that this SR is not required for plants which use the CST under these conditions. This is an ITS Category (iv) change. Acceptable

vi. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including providing consistency with the accident analyses.

b. Various wording changes were made to improve the readability and understanding of the bases.

[ITS80.vi.a and b]:

3.7Q72 As noted in ITS80.i the LCO statement, Applicability and Conditions are in need of rewrite, so no comment will be made until the formatting issues are finalized.

Status: [] Open

Response: See response to 3.7Q62.

3.7Q73

The BASES for SR 3.7.5.2 per change ITS80.iii, last paragraph, does not need to delete last sentence. Also the "may not have been" is

not required and the "are" should be retained.
Status: [] Open
Response: *RG&E agrees to retain the last sentence of the NUREG bases for SR 3.7.5.2. However, the use of "are" for the preceding sentence is not grammatically correct. Instead, RG&E proposes to replace "may not have been" with "have been." Comment #93 has been opened to address this.*

vii. Incorporation of approved Traveller NRC-01, C.2 Acceptable

viii. Incorporation of approved Traveller NRC-13, C.1.

[ITS80.viii]:
3.7Q74 The incorporation of this traveller does not match the NUREG-1431 Revision 1 text in the BASES for SR 3.7.5.2. Please explain difference?

Status: [] Open
Response: *Revision 1 of the NUREG revised the bases text for SR 3.7.5.2 beyond that documented in Traveller NRC-13, C.1 (see attached traveller). The difference in text between that in the Traveller and Revision 1 is minor (i.e., no difference in intent) such that RG&E does not believe that additional bases changes are necessary.*

ix. The Completion Time limit of "10 days from the discovery to failure to meet the LCO" was not added to the new specification since Ginna Station currently does not have this requirement. The intent of adding this limit to the Completion Time is to prevent a plant from continuously being in the LCO without ever meeting the full AFW System requirements. This abuse of the LCO is best addressed under plant procedures since the addition of this limit to the Completion Time creates confusion among licensed personnel. Providing this limit can still result in LCO abuse since the AFW System can be declared OPERABLE for only a several minutes and then the LCO immediately entered for extended periods. Sufficient NRC guidance already exists with respect to extensive use of LCO time (e.g., Ref. 26). In addition, the Maintenance Rule (10 CFR 50.65) requires monitoring of equipment performance. Finally, a review of Ginna Station plant records indicates that the AFW System was out a service a total of 2600 hours from June 1990 and July 1994 (or 9% of the time in which the plant was in MODE 1, 2, and 3.

[ITS80.ix]:
3.7Q75. This Completion time requirement is to guard against "flip-flop" between LCO conditions without restoring equipment fully operable. Since these conditions are to be rewritten per ITS80.i, this issue is deferred because it may not be needed.

Status: [] Open
Response: *See response to 3.7Q62.*

81. ITS 3.7.6

i: The title, LCO, Surveillances and bases were revised to reflect that Ginna Station has two condensate storage tanks (CSTs) instead of one as referenced in NUREG-1431. This is an ITS Category (iv) change.
Acceptable

- ii. The LCO was revised to require that the CSTs be OPERABLE with the specific OPERABILITY requirements specified in the Surveillance Requirement. In addition, Condition A, Required Action A.2, and SR 3.7.6.1 were revised to replace the reference to CST "level" with CST "water volume" which is the actual parameter used in the accident analyses. These changes provide consistency with LCO 3.5.4. These are ITS Category (iii) changes.

[ITS80.ix]
3.7Q76

It appears acceptable to change to "water volume" limits from "water level" limits. However, it is not clear that this LCO could only be in support of AFW and not also SAFW. What prevents the Condensate Test Tank being used for the source of water for SAFW if the Service Water System were unavailable?

Status: []
Response:

Open

The ITS Applicable Safety Analyses bases state that the CST is credited in the safety analysis for all events "which assumes that the preferred AFW System is available immediately following an accident. For any event in which AFW is not required for at least 10 minutes following the accident, the SW System provides the source of cooling water to remove decay heat" (see Insert 3.7.6.2). Since the SAFW System is manually actuated by operators, this system is not available for 10 minutes following an accident such that the non-seismic, non-safety related, CST is not required for this system. In addition, the only source of water to the SAFW Condensate Test Tank is a 1/2 inch line from the main condenser (i.e., there is no piping connected the CST to the SAFW System). Since the SAFW injection paths are 4 inch lines, this supply path is not acceptable to support decay heat removal.

3.7Q77

What other water tanks are available which could be used by operators to meet the operability requirements of water volume for this LCO? Such as the all-volatile-treatment condensate storage tank and others?

Status: []
Response:

Open

The alternate condensate source for the preferred AFW and SAFW pumps is documented in procedure ER-AFW.1 (attached).

3.7Q78

How does refilling of the CSTs figure into the determination of the OPERABILITY? From the fire water systems or the condenser hotwell?

Status: []
Response:

Open

The CSTs must contain a minimum of 22,500 gallons of water. As long as this water is available, the LCO is met, regardless of whether the CST was filled by the fire water system or condenser hotwell. Due to operation and chemistry concerns, the CST is only filled via the condenser hotwell or other acceptable condensate storage tanks. However, the use of fire water systems is an acceptable means for meeting the LCO requirements since fire water is the same as service water which is credited in the accident analysis for long term decay heat removal.

- iii. The Applicability and Required Action B.2 were revised to only require the CSTs to be OPERABLE in MODES 1, 2, and 3 consistent with current Ginna Station TS 3.4.3. The requirement for the CSTs during MODE 4 when the SGs are being relied upon for heat removal is

controlled by new LCO 3.4.6 which specifies required SG level requirements. Due to the wide variety of means of providing decay heat removal in MODE 4 (e.g., AFW, SAFW, MFW, condensate booster pumps), RG&E does not believe that it is necessary to specify CST requirements. This is also discussed in Reference 28. This is an ITS Category (i) change.

[ITS81.iii]:

3.7Q79

UFSAR 10.5.3.1.2 clearly states that AFW is used to maintain steam generator water level during MODE 4 operations as the reactor is being shutdown. Therefore, since CST is the source for AFW this Applicability requirement and the other LCO deleted requirements must be restored and/or rewritten to add provisions for a SAFW train.

Status: []

Open

Response: See response to 3.7Q65.

3.7Q80

Also adhering to the improved TS format requires agreement between LCOs for Applicability, regardless of the possible in-depth capabilities of various supporting systems.

Status: []

Open

Response: See response to 3.7Q65.

- iv. The Completion Time for Required Action A.1 was revised to remove the continued verification every 12 hours of the backup water supply to the CSTs. The current Ginna Station TS 3.4.3 does not contain this requirement. In addition, the sources of water which would normally be used include the SW System (which has Lake Ontario as a water supply) and the all-volatile-treatment condensate storage tank which has a normal stored volume of 100,000 gallons (UFSAR Section 10.7.4). Either of these sources provide much more water than is required for AFW during a DBA or normal cool down. This is an ITS Category (i) change.

[ITS81.iv]:

3.7Q81

What is the hardship created by this verification of the availability and determination of the OPERABILITY of alternate sources of water? The amount of water available is not as important as there being an OPERABLE pathway to deliver the water.

Status: []

Open

Response: *While the hardship required by this verification is limited, Ginna Station has an emergency procedure outlining the specific actions to be taken to use an alternate suction source for the AFW pumps (attached). The sources of water include the condensate hotwell which must have at least 22,500 gallons of water to support normal operation requirements such that a verification is unnecessary.*

- v. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Plant-specific design considerations were added including providing consistency with the accident analyses and bases for the CST water volume requirement.
 - b. Various wording changes were made to improve the readability

and understanding of the bases.

[ITS81.v.a and b]:

3.7Q82 Background first paragraph, last sentence is deleted; however, UFSAR 10.5.3.1.1 states all pumps have recirculation lines back to the CST?

Status: [] Open

Response: *While all three preferred AFW pumps have recirculation lines back to the CST, these recirculation lines are not providing "continuous recirculation" as the NUREG Background bases state. In addition, this recirculation path is not an accident analysis assumption since the MSSVs prevent deadheading of the AFW pumps. Consequently, this bases text was deleted.*

3.7Q83 Insert 3.7.6.1 first sentence refers to a non-seismic grade Service building. Insert "grade"?

Status: [] Open

Response: *No, RG&E nomenclature does not use "grade" with respect to seismic design, only with respect to "safety."*

3.7Q84 Insert 3.7.6.2, it is presumed that SW water is the preferred (only?) source of water for SAFW regardless of the event or time into event.

Status: [] Open

Response: *SW is the source of water credited in the accident analyses for the SAFW System as stated in Insert 3.7.5.6 for the bases for LCO 3.7.5. Other sources of water are available (see procedure ER-AFW.1) but are not credited.*

3.7Q85 The last sentence of insert 3.7.6.5 is not understood.

Status: [] Open

Response: *The water inventory requirement for the CST is based on a station blackout (SBO) which is a beyond DBA event (i.e., a SBO requires a complete loss of offsite power and failure of both diesel generators). The TMI commitments related to SBO require 2 hours of decay heat removal while the accident analyses only require 10 minutes until the SW System can be used. Since the LCO continues to require the 2 hours of decay heat removal, the water inventory requirement is based on a beyond DBA event.*

3.7Q86 The loss of "all AC" electrical power means "onsite and offsite?" Please change to clarify.

Status: [] Open

Response: *Correct, loss of "all AC electrical power" refers to both onsite and offsite (see response to 3.7Q85). RG&E agrees to clarify this statement. Comment #93 has been opened to address this.*

3.7Q87 In Applicability, MODE 4 is applicable.

Status: [] Open

Response: *See response to 3.7Q65.*

82. ITS 3.7.7

- i. The LCO, Conditions, and bases were revised to reflect the actual design of the component cooling water (CCW) system at Ginna Station.

The CCW System is comprised of two 100% capacity pumps which feed a common loop header. This common loop header then splits into parallel flowpaths for two 100% capacity heat exchangers. The outlet of the heat exchangers then meet to re-form the common loop header which provides cooling water to the safety and nonsafety related system loads. The discharge flow through these system loads then combine to re-form the common header which provides suction to the two CCW pumps. As such, the LCO was revised to require the two CCW pump trains and the CCW loop header to be OPERABLE. The Note for Required Action A.1 was also not added since the inoperability of a single CCW train does not affect the ability of CCW to provide cooling to either RHR heat exchanger. This is an ITS Category (i) change.

[ITS82.i]:

3.7Q88

It is acceptable to reformat the LCO and to not include the note for Required Action A.1 because one common header supplies the heat loads.

Status: [] Closed

Response: N/A

3.7Q89

The existing TS require both CCW heat exchangers (HX) OPERABLE. Isn't this a relaxation to treat one as a standby? Provide justification as such.

Status: [] Open

Response: *This change is discussed in Attachment A, Section D, item 13.xvi.*

3.7Q90

Can one CCW HX handle all the heat load from all the safety-related and non-safety-related components cooled by CCW?

Status: [] Open

Response: *One CCW heat exchanger is normally in service during power operation with the second heat exchanger essentially valved out of service (i.e., the CCW flowpath is open but SW cooling supply is left only slightly open to assist in opening the valves during accident conditions). However, only one heat exchanger is required for accident conditions.*

3.7Q91

How often are the standby components put into service and operation? Are the previous components in service put into standby?

Status: [] Open

Response: *The two CCW pumps are rotated on a monthly basis (only one pump is required for normal operation). The CCW heat exchangers are typically not swapped until the next refueling outage.*

3.7Q92

Why not continue with both HX inservice and have the extra capacity? With the passive component, there should be little maintenance required.

Status: [] Open

Response: *RG&E is not proposing to eliminate the redundant heat exchanger from the system, only revise the LCO requirements. The purpose of the technical specifications is to maintain all accident analysis assumptions. Since the accident analyses do not require the second heat exchanger, and the CCW System is based on the use of a single loop header, the LCO should not require two heat exchangers. [This response was revised during meetings the week of 11/1/95. See*

comment #189.]

- ii. A new Condition was added in the event that both CCW trains or the CCW loop header were inoperable. In this condition, CCW cannot support the OPERABILITY of the ECCS and CS pumps and a loss of multiple safety functions exist. However, it is not prudent to enter LCO 3.0.3 in this condition since it would require entry into MODE 5 where CCW must be available to support the RHR heat exchangers. Instead, the new Condition requires immediate action to restore one CCW train or the loop header and to place the plant in MODE 4 within 12 hours. Restricting the cooldown to MODE 4 places the plant in a condition in which the RCPs and AFW can be used to provide decay heat removal while attempts to restore CCW continue. In the event that the RCPs or AFW is also lost, the time required before RHR must be available for decay heat removal is increased in this lower MODE. The change is also consistent with the Required Actions for a loss of RHR and current Ginna Station TS 3.3.3. This is an ITS Category (i) change.

[ITS82.ii]:

3.7Q93 Why is this not an Owners Group Traveler for this requested change?

Status: [] Open

Response: *The WOG rejected a Traveller on this issue since no other plants currently have this type of CTS requirement. Instead, licensees claimed they would request enforcement discretion if all CCW were lost. RG&E could not justify elimination of this shutdown restriction and therefore maintained the CTS requirement.*

3.7Q94 Shouldn't this Condition also have the note deleted from Required Action A.1?

Status: [] Open

Response: *See response to 3.7Q95.*

3.7Q95 Explain how Ginna will handle this new proposed Condition C with the existence of NUREG-1431 LCO 3.0.6 and Administrative Controls of 5.5.15, Safety Function Determination Program and Condition C of LCO 3.7.8?

Status: [] Open

Response: *RG&E proposes to add a Note similar to that for Required Action D.1 of NUREG LCO 3.7.5. This Note would suspend implementation of LCO 3.0.3 and all MODE reductions until one CCW pump and the loop header are restored to OPERABLE status. Comment #100 has been opened to address this.*

- iii. The CCW System is only required by the accident analyses during the recirculation phase following a LOCA and is manually initiated. Therefore, the CCW System does not receive any actuation signal such that SR 3.7.7.2 and SR 3.7.7.3 are not applicable to Ginna Station. However, a new Surveillance was added to require a complete cycle of the normally closed motor operated valves to the RHR heat exchangers. All other CCW flow paths to components required following a DBA are normally open and do not require testing. This is an ITS Category (i) change. As such, approved Traveller NRC-01, C.2 was not incorporated.

[ITS82.iii]:

3.7Q96

Please clarify the actual intended use for CCW which the BASES state is for normal and accident conditions versus here where CCW is only required by accident analyses (above ITS82.ii imply CCW is essential to the RHR HX in MODE 5).

Status: [] Open

Response: *CCW is essential to the RHR heat exchangers in MODE 5 and 6; however the MODE of Applicability for this LCO is MODES 1, 2, 3, and 4. Consequently, the bases and change justifications only relate to the function served by CCW above MODE 5.*

3.7Q97

Regarding not adding SRs 3.7.7.2 and 3, doesn't the standby CCW pump get a start signal when system pressure drops below 50 psig. Why is this not verified at refueling? Doesn't certain valves have to automatically open to align this pump to the loop header?

Status: [] Open

Response: *The standby CCW does receive a start signal when system pressure drops below 50 psig. However, this start signal is not credited nor required in the accident analyses. This 50 psig start signal is for normal operational concerns only. The CCW System is only required post accident during the recirculation phase at Ginna (i.e., following an SI and UV signal on the 480V buses supplying the CCW pumps, the CCW pumps are stripped and must be manually started prior to implementing recirculation). The 50 psig start signal is currently tested every 18 months. No valves in the CCW System must open or close when the second CCW pump starts (i.e., this pump is always aligned as if in service unless it is in maintenance and the LCO is entered).*

3.7Q98

Traveler NRC-01, C.2 depends on above responses.

Status: [] Open

Response: *See response to 3.7Q97.*

iv. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific design considerations were added including providing consistency with the accident analyses for operation of the CCW System.
- b. Various wording changes were made to improve the readability and understanding of the bases.
- c. The text was revised to provide consistency with the bases for LCO 3.7.8.

[ITS82.iv.a, b and c]:

3.7Q99

Provide a sketch of the loop header from the first and last isolation valves for each safety-related load. What type of valves are these? Which SR's apply to them? Are all non-essential loads manually or automatically isolated?

Status: [] Open

Response: *A sketch of the CCW System has been provided as requested showing the breakdown of the loop header boundaries. All non-essential CCW loads remain supplied by CCW following an accident (i.e., there is*



no manual or automatic isolation of these loads) except for the line to the reactor support coolers which is isolated by a containment isolation signal. These automatic containment isolation valves are tested by ITS SR 3.6.3.3. All loads required following an accident relate to cooling for the SI, CS, and RHR pumps and the RHR heat exchangers. CCW flow is continuously maintained through the pump loads while the RHR heat exchangers are isolated by two motor operated isolation valves. The two RHR heat exchanger MOVs are tested by ITS SR 3.7.7.2. All valves related to the SI, CS, and RHR pumps are either motor-operated valves or manual valves which are all locked opened such that a surveillance is not required.

3.7Q100 What are the normal CCW water temperatures during plant shutdown?
Status: [] Open
Response: *CCW is typically maintained approximately 85°F during shutdown conditions. However, the MODE of Applicability for the LCO is only MODES 1, 2, 3, and 4.*

3.7Q101 Why delete last sentence of background after insert 3.7.7.3?
Status: [] Open
Response: *The MODE of Applicability of this LCO is MODES 1, 2, 3, and 4. The use of CCW to support RHR during normal cooldown and shutdown is outside this Applicability. The use of CCW to support RHR during post accident cooldown and shutdown is described in Insert 3.7.7.3.*

3.7Q102 The LCO operability requirements are very complex as described. It appears that some information may be relocated to the Background or elsewhere. Perhaps use of a table would help clarify presentation.
Status: [] Open
Response: *The Operations department does not like LCO OPERABILITY requirements located in the Background or any other bases section other than the LCO. RG&E is planning on providing a drawing in the bases which shows the breakdown of the CCW pump trains and loop header for greater clarity. Comment #101 has been opened to address this.*

3.7Q103 The last sentence of insert 3.7.7.5 needs a more explanation.
Status: [] Open
Response: *The radiation detector in the CCW System is not required to be OPERABLE for this LCO since it is not credited in any accident analysis. The purpose of this detector is to support detection of any leaks between CCW and a system containing radioactive fluid during normal operation. However, if the CCW System were credited as a closed system outside containment with respect to containment integrity, this radiation detector would be required to be OPERABLE to automatically isolate the surge tank in the event of an accident. Since the CCW lines inside containment are a closed system, this radiation detector is not required.*

3.7Q104 In Applicability, the LCOs supported by CCW needs identification by name rather than just number.
Status: [] Open
Response: *Consistent with other NUREG bases, if an LCO is discussed multiple times in the bases, the first mention must include the LCO title. However, all other mentions of the LCO do not require their title. The titles of the LCOs discussed in the Applicability bases are*

presented in the LCO bases (see Insert 3.7.7.5).

3.7Q105 Changes made to Conditions and SRs noted above need modifications in BASES markup.

Status: [] Open

Response: See response to 3.7Q88 through 3.7Q104.

3.7Q106 What condition is entered if the surge tank is inoperable?

Status: [] Open

Response: The first sentence of Insert 3.7.7.5 states "the CCW loop header is considered OPERABLE when the associated piping, valves, one of two CCW heat exchangers, surge tank..." Therefore, Condition C is entered.

3.7Q107 Why relocate the note to SR 3.7.7.1 to the end?

Status: [] Open

Response: Throughout the NUREG bases, any Notes to the LCO, Actions, or Surveillances are discussed in the last paragraph of that section. Relocating this Note discussion to the end of SR 3.7.7.1 provides consistency with the rest of the ITS bases.

- v. SR 3.7.7.1 was revised to only require verification of manual and power operated valves in the CCW train or loop header flowpath that service post-accident related equipment which is a more accurate description of the actual SR. This is an ITS Category (iv) change.

[ITS82.v]:

3.7Q108 The same sketch as noted in ITS82.iv, item #1 above and the responses are needed to evaluate this proposed change. The addition of SR 3.7.7.2 seems to counter this justification. The name change from safety-related to post-accident was not explained or justified.

Status: [] Open

Response: The requested sketch has been provided. SR 3.7.7.2 is the verification that the two CCW motor operated valves to the RHR heat exchangers are stroked open and closed in accordance with the IST Program. These two valves are normally maintained closed and opened by operators prior to implementing the recirculation phase as stated in the bases for SR 3.7.7.2. Therefore, SR 3.7.7.2 ensures that these valves can be opened when required. All other valves which serve post-accident components are locked in position or have CCW flow through them during power operation and must be verified by SR 3.7.7.1. The change from "safety related" to "post accident" was made since all heat exchangers supplied by CCW are identified as safety class 3 since nothing on the CCW System is isolated following an accident. Therefore, this change provides greater operator clarity.

- vi. Incorporation of approved Traveller WOG-12, C.3. Not checked

83. ITS 3.7.8

- i. The title was revised to be consistent with Ginna Station nomenclature which does not abbreviate the term "system" with respect to the Service Water (SW) System. This is an ITS Category (iv) change. Acceptable

- ii. The LCO, Conditions, and bases were revised to reflect the actual design of the SW system at Ginna Station. The SW System is comprised of two redundant trains. Each train includes two 100% capacity pumps which feed a common loop header. This common loop header provides cooling water to the safety and nonsafety related system loads. As such, the LCO was revised to require the two SW pump trains and the SW loop header to be OPERABLE. The Notes for Required Action A.1 were also not added since the inoperability of a single SW train does not affect the ability of SW to provide cooling to either RHR heat exchanger or the diesel generators. In addition, the LCO bases were revised to state that the SW loop header ends at the first isolation valve for any supplied component. If cooling water through or from any component required by the accident analysis is unavailable, then the applicable LCO should be entered. This is an ITS Category (i) change. A such, approved Traveller WOG-12, C11 was not incorporated.

[ITS83.ii]:

3.7Q109 It is acceptable to reflect the Ginna SW system design. Also explain what is the normal and standby discharge header noted in UFSAR 9.2.1.1?

Status: [] Open

Response: *All SW is discharged into a common canal where it is directed to Lake Ontario. However, the discharge canal can also be directed towards the Screenhouse water bay which provides the suction source for the SW, fire water pumps, and circulating water pumps. This path can be utilized during winter months to prevent the formation of ice; however, it is normally maintained closed by use of a motor operated valve.*

3.7Q110 It is acceptable to delete notes 1 and 2 to Required Action A.1 provided assurance is established that the cross connect valves will never be closed. How will this be checked, if not by an unique SR?

Status: [] Open

Response: *The verification of the cross connect valves is required by SR 3.7.8.1 (see ballooned text at bottom of Attachment D page B 3.7-43.*

3.7Q111 The BASES need help and rewriting as noted in ITS83.iv.

Status: [] Open

Response: *See responses to 3.7Q119 through 3.7Q125.*

3.7Q112 The seventh sentence above is acceptable. Is this first isolation valve of the SW header the same isolation valve of LCO 3.6.3? Please state how this LCO would work with the LCO Note #3 to LCO 3.6.3? Should a similar note be placed here?

Status: [] Open

Response: *The use of the "SW header" in the LCO is intended to require that the common header which supplies all of the system components is OPERABLE. If there is a failure which affects only one or two loads supplied by SW, and does not affect any other supplied loads, then the SW System should not have to be declared inoperable and immediately initiate a shutdown. Instead, the affected loads are declared inoperable and their associated LCO entered. In many instances, the containment isolation valve addressed in LCO 3.6.3 is*

the boundary with respect to the SW header. If this containment isolation valve was required to be closed, it does not affect the SW header. Instead, the load supplied by SW and the closed valves (e.g., CNMT fan cooler) is declared inoperable and that LCO is entered. While there is some merit to adding a note similar to Note #3 of LCO 3.6.3, this may cause confusion since the intent of the bases is that these supplied loads are not addressed in LCO 3.7.8, only in their applicable LCOs. Maybe the other SW supplied LCO bases need clarification with respect to this. Suggest this be discussed during the Ginna site visit.

3.7Q113 The eighth sentence is correct and the deleted Notes 1 and 2 of Required Action A.1 permit a cascade of these critical safety systems when cooling water was unavailable. In the new proposed Condition C, shouldn't those deleted notes should be added here?

Status:[]

Open

Response: See response to 3.7Q118.

3.7Q114 In sentence ten, the traveler as noted has no comment number as identified as being applicable for this LCO. Please explain.

Status:[]

Open

Response: This is a typographical error in Attachment A. The Traveller in question is actually "WOG-12, C.1." Comment #92 has been opened to correct this.

iii. A new Condition was added in the event that both SW trains or the SW loop header were inoperable. In this condition, SW cannot support the OPERABILITY of the SI pumps, CRFCs, CCW heat exchangers, diesel generators, or AFW pumps and a loss of multiple safety functions exist. However, it is not prudent to enter LCO 3.0.3 in this condition since it would require entry into MODE 5 where SW must be available to support the RHR heat exchangers. Instead, the new Condition requires immediate action to restore one SW train or the loop header and to place the plant in MODE 4 within 12 hours. Restricting the cooldown to MODE 4 places the plant in a condition in which the RCPs and AFW can be used to provide decay heat removal while attempts to restore SW continue. In the event that the RCPs or AFW is also lost, the time required before RHR must be available for decay heat removal is increased in this lower MODE. This is an ITS Category (i) change.

[ITS83.iii]:

3.7Q115 Why is this not an Owners Group Traveler for this requested change?

Status:[]

Open

Response: The WOG rejected a Traveller on this issue since no other plants currently have this type of CTS requirement. Instead, licensees claimed they would request enforcement discretion if all SW were lost. RG&E could not justify elimination of this shutdown restriction for CCW. Since the loss of all SW would also result in the loss of CCW, RG&E maintained the CTS requirement.

3.7Q116 The acceptance of this Condition is dependent upon response to item #5 of the above [ITS83.ii].

Status:[]

Open

Response: See response to 3.7Q118.

3.7Q117 Given the stated importance of the SW supported systems, shouldn't all the supported systems of SW system be cascaded in this LCO?
Status: [] Open
Response: See response to 3.7Q118. Also, the SFDP requires a type of cascading if LCO 3.7.8 is entered.

3.7Q118 Explain how Ginna will handle this new proposed Condition C with the existence of NUREG-1431 LCO 3.0.6 and Administrative Controls of 5.5.15, Safety Function Determination Program?
Status: [] Open
Response: RG&E proposes to add a Note similar to that for Required Action D.1 of NUREG LCO 3.7.5. This Note would suspend implementation of LCO 3.0.3 and all MODE reductions until one SW pump and the loop header are restored to OPERABLE status. Comment #102 has been opened to address this.

- iv. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Plant-specific design considerations were added including providing consistency with the accident analyses for operation of the SW System.
 - b. Various wording changes were made to improve the readability and understanding of the bases.
 - c. The text was revised to provide consistency with the bases for LCO 3.7.7.

[ITS83.iv.a, b and c]:

3.7Q119 As noted above, it is acceptable to state the Ginna design but this background markup for paragraphs 2 and 3 is illegible. What happened to insert 3.7.8.1?
Status: [] Open
Response: A new copy of the first page of the bases for LCO 3.7.8 is being provided. Insert 3.7.8.1 is used in the LCO to add new Condition C.

3.7Q120 With the addition of insert 3.7.8.4, there is redundant wording in the markup.
Status: [] Open
Response: The first two sentences of the ITS bases for Applicable Safety Analysis (where Insert 3.7.8.4 was placed) now read:
"The design basis of the SW System is for one SW train in conjunction with a 100% capacity containment cooling system (i.e., containment recirculation fan coolers) to provide for heat removal following a steam line break (SLB) inside containment to ensure containment integrity. The SW System is also designed, in conjunction with the CCW System and a 100% capacity Emergency Core Cooling System and containment cooling system, to remove the loss of coolant (LOCA) heat load from the containment sump during the recirculation phase."
There is no redundant wording in these sentences since the first sentence relates to SW support of containment integrity while the second sentence relates to SW support of containment cooling during recirculation.



3.7Q121 In insert 3.7.8.2.a, the acceptable and unacceptable electrical train pairs should be stated. The electrical/mechanical division need better descriptions.

Status:[] Open

Response: *The acceptable and unacceptable electrical train pairs is discussed in the LCO bases (see inserted text into second paragraph of LCO bases on Attachment D page B 3.7-42).*

3.7Q122 In the September 5th telecon, it was discussed that the addition of the Ultimate Heat Sink should be placed in its own LCO as per the guidance of NUREG-1431. Therefore, the contents of the LCO operability BASES could be significantly reduced in size to make a more concise statement. Various parts are background and could be relocated as appropriate. Suggest use of a table to clarify operability requirements.

Status:[] Open

Response: *See response to 3.7Q129. In addition, the UHS related issues discussed in the LCO bases only comprise 3 sentences of a 3 page discussion which is not significant. Also, the Operations department does not like LCO OPERABILITY requirements located in the Background or any other bases section other than the LCO. RG&E is planning on providing a drawing in the bases which shows the breakdown of the SW pump trains and loop header for greater clarity. Comment #103 has been opened to address this. [See also comment #142]*

3.7Q123 In Applicability, the LCO supported should be identified by name rather than numbers due to renumbering possibilities.

Status:[] Open

Response: *Consistent with other NUREG bases, if an LCO is discussed multiple times in the bases, the first mention must include the LCO title. However, all other mentions of the LCO do not require their title. The titles of the LCOs discussed in the Applicability bases are presented in the LCO bases (see Insert 3.7.8.9).*

3.7Q124 Insert 3.7.8.11 may change based upon comments to ITS83iii.

Status:[] Open

Response: *See response to 3.7Q118.*

3.7Q125 Why move the note description to SR 3.7.8.1? The six-sets of isolation valves need identification. Also, how are the cross connect valves defined are separate from others and why shouldn't they be verified as open in their own SR?

Status:[] Open

Response: *Throughout the NUREG bases, any Notes to the LCO, Actions, or Surveillances are discussed in the last paragraph of that section. Relocating this Note discussion to the end of SR 3.7.8.1 provides consistency with the rest of the ITS bases. The six sets of isolation valves and cross-connects are identified in the LCO bases (see Insert 3.7.8.9). Having a separate SR for different valves when it is the exact same surveillance requirement for all valves only creates confusion and is probably not necessary.*

v. Incorporation of approved Traveller WOG-12, C.3.

[ITS83.v]:

3.7Q126

Explain where and why this was incorporated into this LCO. The Excel listing shows this traveler was limited to LCO 3.7.7!

Status:[] Open

Response: *This is a typographical error in Attachment A and should be deleted. Comment #92 has been opened to replace Change 83.v with "Not used."*

vi. Incorporation of approved Traveller NRC-01, C.2. Acceptable

84. ITS 3.7.9

i. This LCO and associated bases were not added to the new specifications. The current Ginna Station TS do not contain any requirements for the Ultimate Heat Sink (UHS). In addition, the UHS for Ginna Station is Lake Ontario (there are no installed cooling towers) and the only safety related function which requires the UHS is the SW System. As such, the bases for LCO 3.7.8 were revised to specify that the SW trains are considered OPERABLE when sufficient NPSH is available and the temperature of the SW suction source was within acceptable limits. These limits are then controlled by the Bases Control Program. It should be noted that the NPSH requirement for the SW pumps is far less than other equipment (i.e., Circulating Water Pumps and Fire Water Pumps) such that sufficient alarms and indications would be available to plant operators. This is an ITS Category (i) change.

[ITS84.i]:

3.7Q127

How is Ginna different from other Westinghouse standard designs or other plants which take their water from lakes or cooling ponds?

Status:[] Open

Response: *The Ginna design with respect to use of lake water is similar to other Westinghouse standard designs.*

3.7Q128

Doesn't the UHS still satisfy the Criterion #3 of the NRC Policy statement?

Status:[] Open

Response: *Yes, the UHS does satisfy Criterion #3 but is not contained in the CTS, nor in the TS of other older plants which use lake water (e.g., Fitzpatrick and Zion Station which is not proposing to add this LCO). RG&E had proposed to relocate this information to the bases for SW OPERABILITY such that the Bases Control Program would control these limits. The UHS temperature and level limits are controlled by procedure (see attached 0-6.6). [See also comment #142]*

3.7Q129.

The UHS is the source for the SW System so retain this LCO with only Condition B as the "UHS is inoperable." with SRs 3.7.9.1 and 3.7.9.2. The proposed inserts 3.7.8.6 and 3.7.8.9 to LCO 3.7.8 are clearly the basis for needing a separate LCO.

Status:[] Rejected

Response: *RG&E would be willing to add this LCO only if the specific temperature limits of SR 3.7.9.2 could be retained outside of the LCO. As discussed in Insert 3.7.8.9 for LCO 3.7.8, the minimum SW temperature changes with respect to power level at Ginna. This is due to the fact that the minimum SW temperature is only important with respect to the design basis LOCA at 100% RTP. In addition,*

RG&E is currently in the process of revising the upper SW temperature limit based on this past summer's extremely hot conditions.

As a side note, why can't NUREG SR 3.7.9.1 (without the specific temperature limit) and SR 3.7.9.2 just be relocated to the SW LCO. Remember that new LCO 3.7.8 Condition C would prevent a power reduction below MODE 4 with the SW loop header inoperable. Based on the SFDP, with the UHS inoperable, the SW loop header and CCW would also be inoperable. Suggest this be discussed during the Ginna site visit. [See also comment #142]

3.7Q130 Renumbering of LCOs, SRs and BASES are required.
Status: [] Rejected
Response: See response to 3.7Q129.

3.7Q131 Modification of BASES for 3.7.8 is expected.
Status: [] Rejected
Response: See response to 3.7Q129.

85. ITS 3.7.10

- i. The LCO title was renamed consistent with Ginna Station nomenclature. In addition, the LCO was renumbered due to the deletion of LCO 3.7.9 (UHS). These are Ginna TS Category (iv) changes.

[ITS85.i]:
3.7Q132 It is acceptable to rename the title of this LCO.
Status: [] Closed
Response: N/A

3.7Q133 The renumbering is not required because there will be no deletion of LCO for UHS. This will only be noted once in spite of other renumbering.
Status: [] Open
Response: See response to 3.7Q129.

- ii. The Control Room Emergency Air Treatment System (CREATS) consists of one filtration train and redundant dampers (see bases and current Ginna Station TS 3.3.5). The current Ginna Station TS allow the filtration train to be inoperable up to 48 hours since the successful operation of the control room isolation dampers will result in acceptable doses within the control room. However, if any radioactive gas were released and entered the control room environment, there is no means to remove the gas. Therefore, 48 hours was determined to be acceptable Completion Time for restoring the system to OPERABLE status or to place the CREATS in the toxic gas mode. In addition, since there are redundant dampers, inoperability of the dampers were treated similar to the CREF trains in NUREG-1431 (i.e., a Completion Time of 7 days is allowed to restore one inoperable damper and a requirement to enter LCO 3.0.3 with two inoperable dampers for a given outside air flowpath). These changes provide consistency with the accident analyses and with NUREG-1431 to the greatest degree possible. These are ITS Category (i) changes. As such, approved Travellers WOG-24, C.5 and

NRC-01, C.2 were not incorporated.

[ITS85.ii]:
3.7Q134

With the filtration train inoperable, CREATS is inoperable as a cleanup system. Condition A should just state "CREATS is inoperable". This is the way existing TS 3.3.5.2 is stated. The insert 3.7.9.2 is acceptable if the logical connector were AND therefore no gas radioactive or toxic could enter the control room as justified above. The note to new A.2 defeats this required action. The time should be limited to 5 minutes or less for one room air exchange with the operators on Scott Air packs.

Status: []

Open

Response:

The reason that Condition A states "CREATS filtration train" is that Conditions B, E, and F address the failure of CREATS dampers. Together, the filtration train and dampers comprise the CREATS such that Condition A should remain as proposed (see attached sketch which is being proposed to be added to the bases). CTS 3.3.5.2 does not address the CREATS dampers which is a "hole" that is being proposed to be fixed during the conversion (see Change D.13.xxii). The Note to new Required Action A.2 is consistent with CTS 3.5.6.2 which allows the control room to be unisolated for 1 hour every 24 hours with the CREATS actuation instrumentation inoperable. RG&E is not willing to revise this CTS allowance nor require operators to wear Scott Air packs during this time frame since the CREATS is comprised of only a single train. RG&E also does not believe that the proposed logical connector to Insert 3.7.9.2 is required. The toxic gas monitors do not meet any of the four criteria for inclusion within technical specifications and have been relocated to the TRM. Including the logical connector will only create confusion for operators since the toxic gas monitors are not in the ITS and all other CREATS instrumentation is addressed in ITS LCO 3.3.5. Comment #104 has been added to added the sketch to the bases.

3.7Q135

New Condition B is acceptable if worded this way: "One CREATS redundant isolation damper open and inoperable in one or more outside flowpaths". To which redundant dampers or damper pairs does this Condition B apply? With any damper closed and inoperable (example AKD07 or AKD09), full operation of the CREATS is not possible. This is also the same as new Conditions E and F for an unisolable flowpath to the outside which are subsets of CREATS being inoperable or being in Condition A. Please explain or correct if this is not the case. Also a copy of UFSAR Figure 6.4-1 should be provided.

Status: []

Open

Response:

All of the CREATS isolation dampers to the outside are redundant, such that the use of "redundant" in the Condition statement is unnecessary. Also, a CREATS damper could be closed but leaking by such that it does not meet the accident analysis assumptions. This is similar to containment isolation valves which do use "containment isolation valve open and inoperable" in their condition statements. The isolation dampers which require entry into Condition B are AKD10, AKD01, AKD08, AKD05, and AKD04 as shown on the attached sketch. With AKD07 or AKD09 inoperable, the CREATS filtration train is declared inoperable. A copy of UFSAR Figure 6.4-1 is being provided as requested.

3.7Q136 Condition A and B should be reversed for descending Completion Times.

Status:[] Open

Response: *The ITS is organized the same as the NUREG which does not have descending Completion Times for this LCO. Therefore, no change is proposed.*

- .iii. Condition C was revised to require placing the OPERABLE CREATS isolation dampers in the emergency radiation protection mode whenever the Required Actions of Conditions of A or (new) B are not met in MODE 5 or 6, or during fuel movement. The emergency radiation protection mode is a more conservative configuration since no outside air is allowed into the control room. Since the dose rates to the operators has been determined to be acceptable with the control room isolated, this is preferred the configuration. This is an ITS Category (i) change.

[ITS85.iii]:

3.7Q137 New Condition D (revised Condition C) should keep "or during movement of irradiated fuel assemblies".

Status:[] Open

Response: *ITS Condition D applies during movement of irradiated fuel movement, not Condition C. Therefore, ITS Condition C does not need to keep "or during movement of irradiated fuel assemblies."*

3.7Q138 The Logical Connector between D.1 and D.2 should be OR.

Status:[] Open

Response: *This is a typographical error in the Required Actions and bases. Comment #105 has been opened to change this logical connector to an OR.*

3.7Q139 The markup states "Place in Mode F". Is this the same as the "emergency radiation protection mode" noted above? The BASES do not have names associated with the Modes A thru F.

Status:[] Open

Response: *The definition of CREATS Modes A through F is contained in the Background bases (see Insert 3.7.9.4). The "emergency radiation protection mode" is the same as Mode F.*

- iv. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including providing consistency with the accident analyses for operation of the CREATS.

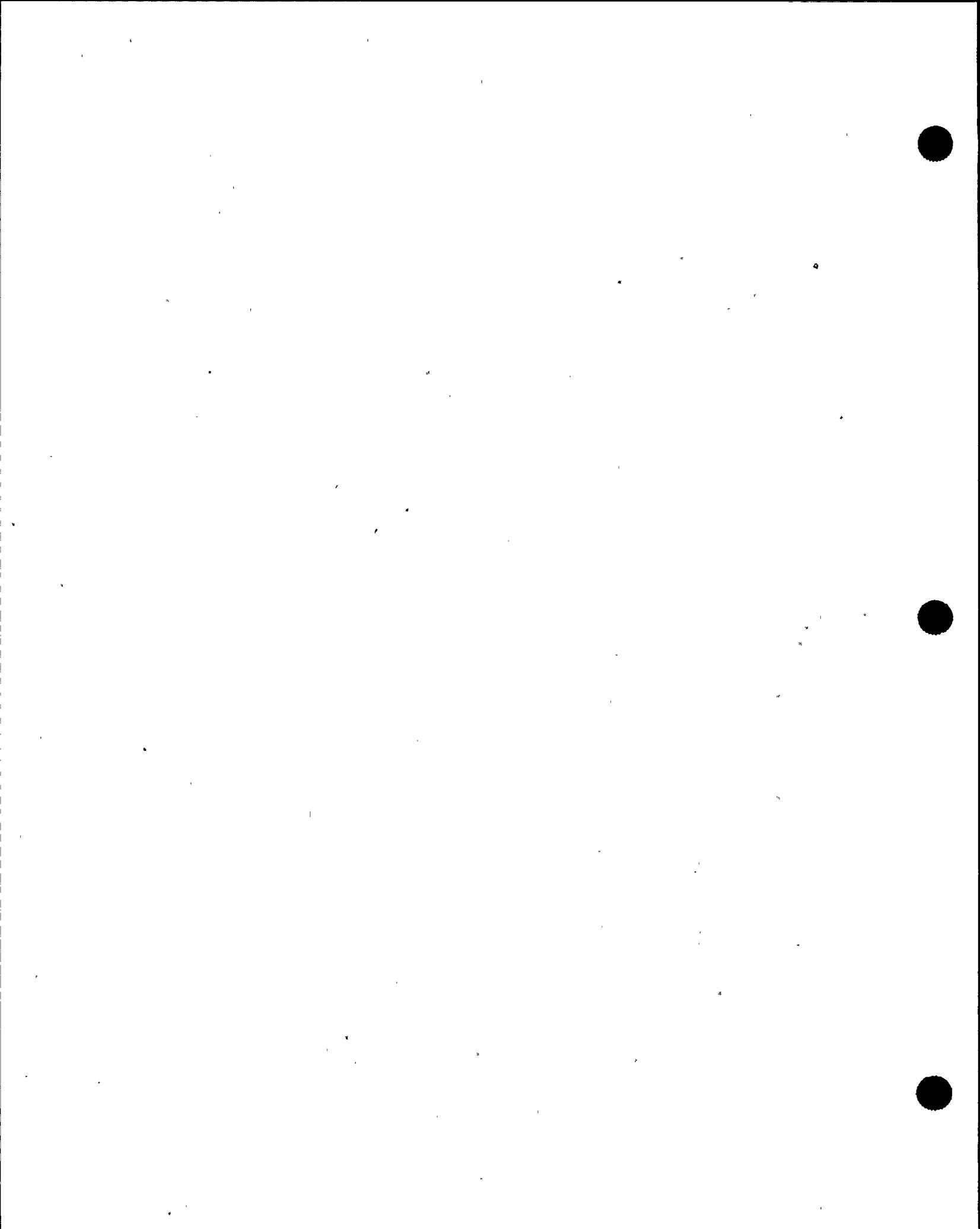
b. Various wording changes were made to improve the readability and understanding of the bases.

[ITS85.iv.a and b]:

3.7Q140 What happened to inserts 3.7.9.1 and 3.7.9.2?

Status:[] Open

Response: *Inserts 3.7.9.1 and 3.7.9.2 are in the LCO section and add new Required Action A.2 and Condition B, respectively.*



- 3.7Q141 In the first paragraph of Background, chemical or toxic gas need not be deleted.
 Status:[] Open
 Response: *The reference to chemical and toxic gas was deleted since this LCO does address the toxic gas requirements of CREATS. These requirements are addressed in the TRM. Ginna operators have requested that the Background bases refrain from discussing items not related to the LCO OPERABILITY requirements. This is also consistent with the ITS Writer's Guide.*
- 3.7Q142 In the second paragraph of Background, CREATS should be stated as one train only. Redundant damper pairs are not mentioned. Also the air conditioning unit and its AC fan are not mentioned.
 Status:[] Open
 Response: *RG&E proposes to add a figure to the bases (see response to 3.7Q134) to provide additional clarity such that the suggested text changes are not required. The air conditioning unit and its associated AC fan are not required for successful operation of the CREATS system and are therefore not discussed in these bases.*
- 3.7Q143 In insert 3.7.9.4, the names associated with the Modes A thru F are not identified as used in revised Condition C or the following paragraph of the Background.
 Status:[] Open
 Response: *The only difference between the names provided in the Background bases and ITS Condition C is that the bases state "CREATS Mode F" while Condition C uses "Mode F." This is an insignificant difference.*
- 3.7Q144 The deletion of paragraph 7 and 9 should be retained and modified for the Ginna design parameter.
 Status:[] Open
 Response: *RG&E agrees to retain NUREG Background bases paragraph 9. Comment #93 has been opened to address this. However, paragraph 7 is not applicable to the Ginna design since CREATS does not create a positive control room pressure. Instead, CREATS is intended to isolate the control room with an equivalent pressure both inside and out.*
- 3.7Q145 The third deleted paragraph of Applicable Safety Analyses should be relocated to LCO to state that any active failure of a component will impair the CREATS operation.
 Status:[] Open
 Response: *The LCO bases already state that the total system failure could result in excessive operator doses and that the CREATS is comprised of "a filtration train." Therefore, RG&E does not believe the deleted third paragraph is necessary.*
- 3.7Q146 In the LCO, the redundant damper pairs should be identified separate from those in insert 3.7.9.7. Also, the last balloon insert to this section is not explained, justified or understood.
 Status:[] Open
 Response: *The redundant outside damper pairs are those listed in Insert 3.7.9.7 (see attached sketch). The balloon insert relates to the fact that the Ginna control room has a single access door (i.e.,*

there is not two doors in-series). The inserted text clarifies that this door can be opened and not violate the LCO. The door can also be opened for an extended period (e.g., to bring equipment in) as long as there is a dedicated individual to isolate and close the door if required.

3.7Q147 In Applicability, why are the waste gas decay tanks located offsite and then, if so, why would they affect the control room. What is [82.vi] justifying here?

Status:[] Open

Response: *The ballooned text "located either offsite or onsite" is a typographical error that is not shown in Attachment C. Comment #93 has been opened to remove this text from Attachment D.*

3.7Q148 Specific questions on the Actions and SRs will be given later after responses to the above LCO questions are received.

Status:[] Open

Response: *See responses to 3.7Q132 through 3.7Q147.*

- v. SR 3.7.10.4 was not added to the new specifications. The current Ginna Station technical specifications do not contain this requirement. The control room environmental control systems were assessed as part of TMI Action Plan requirements (i.e., NUREG-0737, Supplement 1, item III.D.3.4) and found to be acceptable (Ref. 28). Therefore, RG&E does not believe that this surveillance is required. This is an ITS Category (i) change.

[ITS85.v]:

3.7Q149 Reference 28 is not available but the control system is acceptable assuming the system can maintain its function. This SR merely tests to verify that the control room can be pressurized at the designed flow rate to minimize the air infiltration. Without this test, how is CREATS operation determined to be OPERABLE.

Status:[] Open

Response: *The control room dose assessments do not assume that the control room is pressurized. Consequently, NUREG SR 3.7.10.4 is not applicable to the Ginna design. Also, Standard Review Plan 6.4 (attached) does not require the use of pressurized control room volume. It only requires verification of inleakage to the control room if the analyses assume a gross leakage rate of < 0.06 volume changes per hour or if a pressurized system is used that has pressurization rates < 0.25 volume changes per hour. UFSAR Table 6.2.4 shows an unfiltered leakage of 0.06 changes per hour such that this verification is not required.*

3.7Q150 As noted in ITS85.i above, SR 3.7.11.1 is needed here.

Status:[] Open

Response: *See response to 3.7Q152.*

- vi. Approved Traveller WOG-12, C.2 was not added since the CREATS is required to be OPERABLE in MODE 6 which includes all CORE ALTERATIONS by definition. This is an ITS Category (iv) change.

[ITS 85.vi]:

3.7Q151 This appears acceptable. What does the traveler state as the



justification or reason for adding this to this LCO?
Status: [] Open
Response: *The justification for Traveller WOG-12, C.2 is attached. Essentially, this Traveller corrected a discrepancy between the instrumentation and hardware CREATS requirements.*

86. ITS 3.7.11

- i. This LCO and associated bases were not added to the new specifications. The current Ginna Station TS do not contain any temperature control requirements for the control room environment. The existing system was evaluated and found to be acceptable as part of the TMI Action Plan Requirements (i.e., NUREG-0737, Supplement 1, item III.D.3.4). This is an ITS Category (i) change. As such, approved Travellers WOG-12, C.2 and WOG-24, C.1 were not incorporated.

[ITS86.i]:
3.7Q152

NUREG-1431 LCO 3.7.11, CREATCS is now a part of the Ginna proposed LCO 3.7.9. Per UFSAR Section 6.4, the Air Conditioning Unit is an integral in-line series component of CREATS operating flowpath. If this unit is inoperable, CREATS is inoperable. The air conditioning unit is just another component in determining the operability of CREATS, so only the SR 3.7.11.1 needs to be added to the Ginna proposed LCO for CREATS.

Status: [] Open

Response: *Although the control room air handling room is in the path of the CREATS system, it is not required to be OPERABLE to support CREATS operation (i.e., the dose analyses do not credit the use of this air handling unit). Consequently, SR 3.7.11.1 is not required.*

87. ITS 3.7.12

- i. This LCO and associated bases were not added to the new specifications. The current Ginna Station TS do not contain any requirements for a ECCS pump room exhaust air cleanup system (PREACS). The bases for this LCO state that the PREACS is used for filtering air from the area of active ECCS components during the recirculation phase of an accident and to provide environmental control (e.g., temperature and humidity). Standard Review Plan 15.6.5 states that for plants which do not provide an ESF atmosphere filtration system, 50 gpm leakage from a gross failure of a passive component should be assumed 24 hours after an accident. This is assumed for Ginna Station with respect to the RHR pumps (UFSAR, Section 5.4.5.3.5). In addition, UFSAR Section 9.4.2 states that the cooling systems related to ECCS equipment are not required even with both trains of ECCS in operation. Therefore, this LCO does not apply to Ginna Station and was not added to the new specifications. This is an ITS Category (i) change.

[ITS87.i]:
3.7Q153

It should be noted that the LCO provides requirements for both room temperature control and air cleanup systems. The UFSAR reference is assumed to be specifically 9.4.2.4.1, Effect of Loss of Cooling on Pumps and Valves. This does state that all pumps in this room can



operate without cooling systems. The UFSAR reference to Section 5.4.5.3.5 could not be accessed due to an incompatible file error. Please provide this reference to show that this LCO is not required for this pump room.

Status: []

Open

Response:

The requested UFSAR section is attached. In addition, it should be noted that the ECCS room coolers do not have any filtration components associated with them, only fan coolers.

3.7Q154

UFSAR 9.4.9, Engineered Safety Features Ventilation Systems state other rooms and areas of the plant where systems are installed require cooling. Please evaluate these areas per the LCO criterion of the policy statement. Example the Standby Auxiliary Feedwater System pump room appears to be only an addition and may not be a part of the Auxiliary Building Ventilation System.

Status: []

Open

Response:

The only air cleanup systems credited in the Ginna accident analyses are for the control room (ITS LCO 3.7.9), the Containment Post-Accident Charcoal Filters (ITS LCO 3.6.6) and the Spent Fuel Pool portion of the Auxiliary Building Ventilation System (ITS LCO 3.7.10). The only room temperature control systems required to ensure that TS systems are OPERABLE are for the diesel generators (see Insert 3.8.1.3 to LCO bases for ITS LCO 3.8.1) and the SAFW pumps (see inserted text in balloon for LCO bases on Attachment D page B 3.7-46 for ITS LCO 3.7.5).

88. ITS 3.7.13

- i. The LCO title was renamed consistent with Ginna Station nomenclature since there is no separate fuel building. In addition, the LCO was renumbered due to the deletion of previous sections. These are Ginna TS Category (iv) changes. Acceptable
- ii. The LCO, Applicability, Conditions, Surveillances, and the bases were all revised to be consistent with current Ginna Station TS 3.11.1. This requires the Auxiliary Building Ventilation System (ABVS) associated with the SFP to be OPERABLE when fuel is being handled or stored in SFP which has decayed < 60 days since being irradiated. The ABVS is defined as one Auxiliary Building exhaust fan, the Auxiliary Building exhaust fan IC, SFP charcoal absorbers, and roughing filters. The ABVS only ensures that offsite doses are well within 10 CFR 100 limits in the event of a fuel handling accident. If the ABVS were unavailable, offsite doses would increase, but remain below 10 CFR 100 limits. Therefore, single failures or a loss of offsite power is not a consideration for this LCO. If the minimum ABVS is inoperable, the new Condition requires suspension of movement within the Auxiliary Building immediately which prevents a fuel handling accident from occurring that requires the ABVS. Since the ABVS is a non-Engineered Safety Features system, and is only required following a fuel handling accident, the majority of surveillance requirements do not apply to Ginna Station. The only SRs which are necessary are those related to the VFTP and to ensure that the system is in operation during fuel movement or CORE ALTERATIONS (new SR 3.7.10.1). All other SRs which require operation of system heaters (SR 3.7.13.1), verify actuation of the

ABVS on a safety injection signal (SR 3.7.13.3), verify the ability to maintain a negative pressure in the fuel handling building (SR 3.7.13.4) or verify that the bypass damper can be closed (SR 3.7.13.5) do not apply to Ginna Station and were not added. These are ITS Category (i) changes. As such, approved Traveller WOG-24, C.6 was not incorporated.

[ITS88.ii]:

3.7Q155

It is acceptable to adapt the FBACS LCO to the Ginna ABVS design and to place the components required OPERABLE in the BASES. The markup as supplied provides information which confuses the identity of the respective ventilation fans, filters and ductwork in the operating train(s?). Please provide the UFSAR Figures 9.4-4 through 9.4-8 to clarify the intent of insert 3.7.10.1.

Status: []

Open

Response:

The requested UFSAR figures have been provided. However, to assist in your review, a sketch of the ABVS is also provided to show what portion of that system is required for this LCO. RG&E proposes to add this sketch to the bases to provided clarity. Comment #106 has been opened to address this.

3.7Q156

This LCO should also applies other accidents besides just a fuel handling accident. As noted in the justification for deletion of NUREG-1431 LCO 3.7.12, there is no room cleanup system, so radioactive particulates must be filtered by the respective building ventilation systems. There is no mention of the portions of the intermediate building and other auxiliary building areas which are also ventilated by ABVS. Also since the High Energy Line Break Analysis was performed after the plant was designed, what effect would the steam isolation dampers not closing have upon the operation of ABVS. Please explain.

Status: []

Open

Response:

This LCO is only applicable for fuel handling accidents. The ABVS does provide air cleanup during normal power operation, but is unavailable following a loss of offsite power and is not considered safety related (see UFSAR Section 3.11.3.2.1). Therefore, for all radiological accidents other than fuel handling accidents, the ABVS is not credited as allowed by Standard Review Plan 15.6.5, Attachment B. With respect to HELBs, there are no steam isolation dampers at Ginna Station. The HELB of concern in the Auxiliary Building is a non-safety grade heating steam line which was addressed during the Systematic Evaluation Program (see UFSAR Section 3.11.3.2.1).

3.7Q157

ABVS appears to be continuously in operation for reasons other than a just FHA. Why shouldn't the Applicability be "at all times." All Owners Groups have accepted any irradiated fuel movement as appropriate applicability rather than a ≤ 60 day limitation. Why can't Ginna?

Status: []

Open

Response:

See response to 3.7Q156 with respect to Applicability. Also, the NRC has recently issued SERs allowing licensees to change their Applicability to be consistent with Ginna's (see Amendment No. 105 to Millstone 3 license dated February 22, 1995). As such, RG&E is not willing to change the Applicability.

3.7Q158 Since the ABVS is in operation, it is acceptable to replace with new SR 3.7.10.1.

Status:[] Closed

Response: N/A

3.7Q159 It is acceptable (as noted in CTS21.ii) that a negative pressure is now the basis for acceptable operation in lieu of existing TS 3.11.1.c; so, the retention of old SR 3.7.13.4 is appropriate.

Status:[] Closed

Response: N/A

3.7Q160 The ABVS train realigns itself upon a high radiation signal as noted in UFSAR 9.4.2.2.1 and insert 3.7.10.1. Therefore old SRs 3.7.13.3 and 3.7.13.5 should be retained.

Status:[] Open

Response: *As discussed in the response to 3.7Q156, the ABVS is non-safety related and is unavailable following a loss of offsite power. As such, even though the ABVS will attempt to realign itself upon a high radiation signal, this realignment cannot be assumed or credited. The ABVS related to the spent fuel pool does not realign following a high radiation signal although it will also be unavailable following a loss of offsite power as discussed in the LCO bases.*

iii. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including using Ginna Station nomenclature and providing consistency with the accident analyses for operation of the ABVS.

b. Various wording changes were made to improve the readability and understanding of the bases.

[ITS88.iii.a and b]:

3.7Q161 There is no mention of the new fuel area which is served by the air handling unit of ABVS.

Status:[] Open

Response: *Unless the new fuel is irradiated, then this LCO does not apply. Only a fuel handling accident with irradiated fuel has the potential to cause significant offsite doses.*

3.7Q162 In the first paragraph of Background, last sentence, last balloon insert is mislocated.

Status:[] Open

Response: *This is a typographical error as this last balloon insert should be after "humidity in the" portion of this last sentence. Comment #94 has been opened to correct this error.*

3.7Q163 As noted above, insert 3.7.10.1 will need streamlining.

Status:[] Open

Response: *RG&E has proposed to add a figure to the bases (see response 3.7Q155). Any changes to the bases to support this are addressed in Comment #106.*



3.7Q164 In Applicable Safety Analyses, third deleted sentence should stay. In the balloon insert to the fourth sentence, why are the listed components "functional" and not "operable." Lastly, the inserted word just prior to insert 3.7.10.2 is not legible.

Status:[] Open

Response: *The third sentence in the Applicable Safety Analyses is not applicable to Ginna as noted in the responses to 3.7Q154 and 3.7Q156 and should therefore remain deleted. RG&E agrees to replace "functional" with "OPERABLE" in the balloon insert to the fourth sentence. Comment #93 has been opened to address this. The illegible inserted word is "minimum."*

3.7Q165 Insert 3.7.10.3 is acceptable but appears to be best relocated to the background.

Status:[] Open

Response: *RG&E believes Insert 3.7.10.3 is correct where it is since it only replaces the text contained in the first paragraph of the NUREG LCO bases which provides a similar discussion.*

3.7Q166 In LCO, the components operable will be determined after receipt of the Figures accompanying this design, noted above.

Status:[] Open

Response: *See response to 3.7Q155.*

3.7Q167 Specific comments on Applicability, Actions and SRs will come later.

Status:[] Open

Response: *See responses to 3.7Q153 through 3.7Q166.*

iv. The text of SR 3.7.10.2 was revised to reflect that only the SFP Charcoal Absorber System is required to be verified since the ABVS has several charcoal filter components. This is an TS Category (iv) change.

[ITS88.iv]:

3.7Q168 The UFSAR has no mention of the SFP Charcoal Absorber (or Adsorber) System any where when checked by ZYINDEX. What and where is this system?

Status:[] Open

Response: *This system is discussed in UFSAR Sections 3.11.3.2.1 and 9.4.4. The title of this System is consistent with Ginna procedures.*

3.7Q169 It is appropriate that the " ? " system have filter testing per VFTP and an SR is required. Are there other filtering systems which also should be tested as a part of the whole ABVS?

Status:[] Open

Response: *As noted in the response to 3.7Q154 and 3.7Q156, no other portion of the ABVS is required to be tested per the VFTP.*

3.7Q170 The various BASES references do not agree on whether this is an "absorber" filter or an "adsorber" filter. The later is the more likely candidate and matches existing TS 4.11.1.

Status:[] Open

Response: *The use of "adsorber" is correct. Comment #93 has been opened to ensure that all bases references use "adsorber."*

3.7Q171 Existing TS 4.11.1.d is missing for this system and should be added as a new SR. See CTS38.iii.

Status:[] Open

Response: See response to 3.7Q278. As noted above and in Change 38.iii, since the ABVS is required to be in operation and OPERABLE during all movement of fuel which has been irradiated within the last 60 days, but is not required to be OPERABLE at any other time to meet the accident analyses, performing CTS 4.11.1.d on a monthly basis is not necessary. Instead, verification that the system is in operation when being used is all that is required as ITS 3.7.10.1 proposes.

3.7Q172 Why are the Auxiliary Building Charcoal Filters not tested also?

Status:[] Open

Response: See response to 3.7Q154 and 3.7Q156. It should be noted that these filters are normally tested at Ginna, but do not require testing in accordance with technical specifications.

89. ITS 3.7.14

- i. This LCO and associated bases were not added to the new specifications since Ginna Station does not have a penetration room exhaust air cleanup system (PREACS). The bases describe the PREACS as a system which filters air from the penetration area between containment and the Auxiliary Building. At Ginna Station, the containment an Auxiliary Building are joined such that there is no space (i.e., penetration area) between these buildings. Therefore, this LCO is not applicable to Ginna Station. This is an ITS Category (i) change.

[ITS89.i]:

3.7Q173

It is acknowledged that there may not be an area of the plant located as described in the BASES or known as "the penetration area"; however, the LCO is reserved for filtration and/or ventilation systems which perform essentially the same function. Even if there is nothing in the existing TS, Ginna is requested to determine whether a similar function exists for this type of LCO. An example is the UFSAR Section 9.4.1.2.10 refers to a Penetration Cooling System. Is this area also filtered?

Status:[] Open

Response: The Penetration Cooling System is used to cool the concrete around certain containment penetrations and provides no filtration purpose. This system was reviewed during the Systematic Evaluation Program and concluded that it was not required to meet any accident analyses. Instead, this is an operational issue only (i.e., a "good practice"). See response to 3.7Q156 with respect to all other ventilation and filtration systems at Ginna.

90. ITS 3.7.15

- i. The title was revised to be consistent with Ginna Station nomenclature including the use of abbreviation "SFP" for "spent fuel pool." This is an ITS Category (iv) change. Acceptable
- ii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific background information with respect to the design of the spent fuel pool (SFP) and the SFP Cooling System was added.
- b. Discussions of non-TS related functions of maintaining level in the SFP were deleted. This type of information is contained in the UFSAR, procedures, and other more appropriate documents.
- c. Various wording changes were made to improve the readability and understanding of the bases and to reflect plant-specific considerations.
- d. The bases were expanded to discuss why the LCO was not applicable for other plant conditions.

[ITS90.ii.a, b, c, and d]:

3.7Q174 The deleted portion of the first paragraph of Background are true statements. Why removed them? At the end of the balloon insert, begin "It also shields....etc".

Status:[] Open

Response: *The deleted portion of the first paragraph of Background are true statements but unrelated to the LCO requirements. That is, these sentences refer to auxiliary functions of SFP level which are not credited in the accident analyses. Ginna operators have requested that this level of information be removed from the bases to reduce potential confusion as to why the LCO exists. This level of information is best addressed in training.*

3.7Q175 In the first line of Applicable Safety Analyses, pool should be deleted.

Status:[] Open

Response: *This is a typographical error. Comment #92 has been opened to remove "pool" from this first sentence.*

3.7Q176 In insert 3.7.1.5.2, the "decontamination factor" is not understood. Please explain.

Status:[] Open

Response: *The text for Insert 3.7.1.5.2 comes from UFSAR Section 15.7.3.3 and Regulatory Guide 1.25 (attached). Essentially, a "decontamination factor of 100" means that 99% of the total iodine released by damaged rods is retained by the pool water.*

3.7Q177 In several places, the use of the word phrase "active fuel" is not clear. Is this off loaded fuel awaiting reloading? Explain or use different term.

Status:[] Open

Response: *"Active fuel" is the portion of the fuel assembly which contains the fuel rods. The fuel assembly naturally extends beyond this "active fuel." Specifying the LCO limit with respect to above the "active fuel" ensures that there is at least 23 feet of water available for iodine removal.*

3.7Q178 In LCO, what is the level of the SFP required when no fuel movement is occurring and fuel is just being stored?

Status:[] Open

Response: The SFP level is normally maintained at a constant level > 23 feet above the fuel whether fuel movement is taking place or not. However, when no fuel movement is occurring, a fuel handling accident cannot occur. Note that the ITS LCO Applicability is the same as the NUREG for a requirement that Ginna currently does not have.

3.7Q179 The addition of the balloon insert after periodically in the SR 3.7.15.1 is dependent upon the answer to item 5 above. Also, the 31 day frequency is on hold.

Status: [] Open

Response: See response to 3.7Q178.

3.7Q180 The last paragraph of insert 3.7.15.4 is challenged as not an acceptable manner to verify SFP water level.

Status: [] Open

Response: Use of the alarms is the only control room indication that operators have to ensure that ≥ 23 feet of water is available. These alarms are both set above the 23 ft LCO limit. Note that Ginna currently does not have this requirement. Unless the reviewer can propose another option which does not require a plant modification, or accepts the proposed method of performing this SR, RG&E will withdraw this SR.

- iii. The Frequency for SR 3.7.15.1 was revised from 7 days to 31 days. Ginna Station currently does not have this Surveillance Requirement. However, consistent with the current surveillance for SFP boron concentration (Ginna Station TS Table 4.1-2, #17), a monthly surveillance of SFP water level is considered adequate due to the design of the SFP as discussed in the bases. This is an ITS Category (i) change.

[ITS90.iii]:

3.7Q181 The current TS surveillance interval for boron concentration appears to be monthly during all operational modes. The proposed ITS interval of seven days is more consistent with the anticipated activity around the SFP since it is concurrent during the movement of irradiated fuel. This 7-day interval was selected as consistent with safe conduct of operations around a fuel storage pool. Most plants have adopted these recommendations along with the improved TS format. Why can't Ginna?

Status: [] Open

Response: The SFP boron concentration can only change if water from an unborated water source is added. WCAP-14181 was recently submitted to the NRC with respect to crediting the use of boron in the SFP at all times to address boraflex issues. This document calculates a frequency of $< 1.0E-06$ /reactor year for this event. As such, RG&E believes that the frequency of 31 days is adequate.

91. ITS 3.7.16

- i. The title was revised to be consistent with Ginna Station nomenclature including the use of "SFP" for "spent fuel pool." This is an ITS Category (iv) change. Acceptable

- ii. The LCO and associated bases were revised to relocate the actual boron concentration limit to the COLR. This change provides consistency with other similar requirements (e.g., ITS 3.9.1). This is an ITS Category (iii) change.

[ITS91.ii]: Please see questions on CTS28.ii.g!

3.7Q182 Why is this not a WOG proposed traveler to the STS?

Status: [] Open/NRC HOLD

Response: *This LCO has not been addressed by the WOG. Comment #107 has been opened to request a WOG traveller at the next meeting in mid-November.*

3.7Q183 When was the last time the boron concentration had to change to warrant it being placed in the COLR?

Status: [] Open/NRC HOLD

Response: *The boron concentration just changed following NRC approval of the new fuel design for Ginna (see attached SER). Though the SFP is normally maintained at 2000 ppm, the recently approved SFP analysis shows that only 300 ppm is required to meet accident analysis assumptions.*

3.7Q184 This change is on hold because it is dependent upon the co-review with the Reviewers for Chapters 3.9, 4.0, COLR and possibly 3.4.

Status: [] Open/NRC HOLD

Response: *To be discussed at meeting.*

iii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific background information with respect to the design of the SFP was added including discussion concerning Integral Fuel Burnable Absorbers as used in Region 1.
- b. Various wording changes were made to improve the readability and understanding of the bases and to use Ginna Station nomenclature.
- c. The text was revised to provide consistency with the bases for LCO 3.7.17. (Note - this is an ITS Category (iii) change.)

[ITS91.iii]:

3.7Q185 For consistency with the markup for LCO 3.7.17, add "into a region" after "assembly" in the last sentence of Background.

Status: [] Open

Response: *RG&E agrees to add "into a region" in the last sentence of Background. Comment #93 has been opened to address this.*

3.7Q186 Page B 3.7-82 is missing from the markup in this Submittal.

Status: [] Open

Response: *The missing bases page has been provided.*

3.7Q187 Insert 3.7.16.4 is not understood for which analysis?

Status: [] Open

Response: *The SFP FHA analysis assumes that each region of the SFP is filled*

with fuel assemblies in an infinite array to conservatively calculate the potential for recriticality and dose releases.

3.7Q188 Insert 3.7.16.5 is acceptable except for the last sentence. This implies the LCO no longer applies and Ginna can continue fuel movement with the boron concentration out of limit. As deleted in the paragraph, this is not allowed as intended under the guidance of NUREG-1431.

Status: []

Open

Response: *The MODE of Applicability for this LCO states (in both the ITS and NUREG:*

"When fuel assemblies are stored in the SFP and a SFP verification has not been performed since the last movement of fuel assemblies in the SFP."

Once this verification has been performed by ITS SR 3.7.13.1 and SR 3.7.13.2, this LCO no longer applies as Insert 3.7.16.5 states. However, Insert 3.7.16.6 states that the verification of SFP boron concentration is "required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished."

3.7Q189 Please explain insert 3.7.16.6, as stating SR "must continue to be performed" when SR interval is proposed to be lengthened to 31 days. Also the balloon insert in lieu of the deleted text as reliant upon plant procedure is not adequate justification. See ITS91.v.below.

Status: []

Open

Response: *Fuel movement occurs throughout a refueling outage which typically last 45 days. Therefore, ITS SR 3.7.12.1 will most likely be required at least twice during refueling outages. The ballooned text is based on the NUREG bases for SR 3.7.15.1 (see top of page B 3.7-80).*

- iv. Required Action A.2.2 was revised to require performance of a SFP verification instead of verifying that a SFP verification had already been performed. If a SFP verification had been performed, then this LCO would not be in effect per the Applicability. This change provides consistency with the intended option to immediately perform the verification when the SFP boron concentration is not within limits. This is an ITS Category (iii) change.

[ITS91.iv]:

3.7Q190 This is acceptable to actually perform the "SFP verification" instead of verifying whether it had been performed. The definition of what constitutes an "SFP verification" is not defined as yet other than a parenthetical reference. Should the LCO or the BASES clarify this definition?

Status: []

Open

Response: *RG&E believes the parenthetical reference to the two SRs which actually perform this verification is adequate since this LCO only relates to SFP boron concentration while ITS LCO 3.7.13 addresses SFP verification.*

- v. The Frequency for SR 3.7.16.1 was revised from 7 days to 31 days consistent with current Ginna Station TS Table 4.1-2, #17. Since the boron concentration is not expected to change rapidly due to the

large volume of water which is available, a monthly verification is considered acceptable. This is an ITS Category (i) change.

[ITS91.v]:
3.7Q191

The current TS surveillance interval for boron concentration appears to be monthly during all operational modes. The proposed ITS interval of seven days is more consistent with the anticipated activity around the SFP since it is currently based upon the last movement of irradiated fuel. This 7-day interval was selected as consistent with safe conduct of operations around a fuel storage pool. Most plants have adopted these recommendations along with the improved TS format. Why can't Ginna?

Status: []
Response:

Open
The SFP boron concentration can only change if water from an unborated water source is added. WCAP-14181 was recently submitted to the NRC with respect to crediting the use of boron in the SFP at all times to address boraflex issues. This document calculates a frequency of $< 1.0E-06$ /reactor year for this event. In addition, the pool boron concentration would have to change from 2000 ppm to 300 ppm without being noticed (level is checked every 7 days). As such, RG&E believes that the frequency of 31 days is adequate.

92. . ITS 3.7.17

- i. The title was revised to be more consistent with the actual LCO since new fuel can be stored in the SFP if the fuel assembly meets the necessary requirements. Also, the abbreviation "SFP" for "spent fuel pool" was used consistent with Ginna Station nomenclature. This is an ITS Category (iii) change. Acceptable
- ii. The LCO and Applicability were revised to provide requirements for both regions of the SFP. This change was required since both Region 1 and Region 2 have limits with respect to the fuel to be stored in addition to the upper U-235 enrichment limit of 5.05 weight percent. As such, Required Actions were necessary if these limits are exceeded. In addition, separate Surveillances were added for each Region to ensure that the limits are met. These limits are consistent with Reference 29. This is an ITS Category (i) change.

[ITS92.ii]:
3.7Q192

Please provide the relevant parts of Reference #29 to verify these new LCO limits.

Status: []
Response:

Open
A copy of the NRC SER which approved this new limits has been provided.

3.7Q193

Add a new item c. to the LCO statement as follows: "Fuel assemblies not meeting a or b above shall be stored in accordance with Specification 4.3.1.1."

Status: []
Response:

Open
RG&E does not believe that the proposed new item c. is necessary. ITS LCO 3.7.13 address all fuel stored in the SFP and places limits on fuel in both Region 1 and 2 while Specification 4.3.1.1 identifies the general limits of fuel stored in the SFP. If a fuel assembly does not meet the requirements for Regions 1 and 2, it

cannot be stored in the SFP by ITS LCO 3.7.13 and Specification 4.3.1.1 has no additional bearing. Note that the NUREG contains this statement since the specific limits on Region 1 are not in this LCO but are in Specification 4.3.1.1. Ginna has proposed to locate all of these requirements in this LCO.

3.7Q194 Add a new item d. to the LCO statement as follows: "The SFP temperature shall be less than 120°F during normal operations and less than 150°F during full core discharge situations." The capacity of the SFP is limited by the heat removal capability of the SFP pooling system as is noted in first paragraph of UFSAR Section 9.1.2. As noted in UFSAR Section 9.1.3.1, this limit is needed to maintain the structural integrity of the SFP.

Status: []

Response: Open
RG&E does not believe that proposed new item d. is necessary. The UFSAR statement is based on superseded concrete codes which required the SFP to be maintained less than 150°F during accident conditions. This code now states that the concrete surface temperature shall not exceed 350°F during accident conditions which is far above the boiling temperature of the SFP (attached). The SFP cooling requirements were justified to be relocated during the development of the improved standard technical specifications.

3.7Q195 The deleted text of SR 3.7.17.1 shall be retained and similarly added to new Proposed 3.7.13.1.

Status: []

Response: Open
Attachment D does not show any text being deleted from NUREG, SR 3.7.17.1, only a change to the numbers and added text.

iii. The bases were revised as follows (these are ITS Category (iv) changes):

- a. Plant-specific background information with respect to the design of the SFP was added including discussion concerning Integral Burnable Absorbers as used in Region 1.
- b. Various wording changes were made to improve the readability and understanding of the bases and to use Ginna Station nomenclature.
- c. The text was revised to provide consistency with the bases for LCO 3.7.16. (Note - this is an ITS Category (iii) change.)

[ITS92.iii.a, b and c]:

3.7Q196 In Background, the last sentence of the first paragraph shall be retained. Also retain this deleted sentence in LCO.

Status: []

Response: Open
See response to 3.7Q193.

3.7Q197 In Background, second paragraph second sentence should be "specify that a limiting k_{eff} ..." to agree with markup to LCO 3.7.16.

Status: []

Response: Open
RG&E agrees to revise the Background bases as proposed. Comment #94

has been opened to address this.

3. 7Q198 New BASES Figure B 3.7.16-1 is missing and is needed for further review.

Status: [] Open

Response: *The missing bases figure has been provided.*

3.7Q199 Other comments on LCO Statement, Conditions, Actions may come depending upon the resolution of ITS92.ii.

Status: [] Open

Response: *See responses to 3.7Q192 through 3.7Q198.*

93. ITS 3.7.18

i. This LCO was renumbered due to the deletion of previous LCO sections. This is an ITS Category (iv) change. Acceptable

ii. The Completion Times for Required Actions A.1 and A.2 were revised consistent with current Ginna Station TS 3.1.4.4 which allow 8 hours to reach MODE 3 and 40 hours to reach MODE 5. This small increase in time is considered minor and acceptable since the accident analyses have been performed using very conservative values for primary system I-131 and no credit is taken for activity plateout or retention. This is an ITS Category (i) change.

[ITS93.ii]:

3.7Q200

Will this be the only place in the improved Ginna TS where the times for an orderly shutdown will be different for all others? If so, why not standardize across all the improved Tech Specs for what has been stated as a minor increase. A small difference like this creates more confusion in procedure writing than will be gained by the extra two hours at each MODE level.

Status: [] Open

Response: *No, ITS LCO 3.4.16, Required Action C.1 also has different shutdown Completion Times than the NUREG. This was accepted by the NRC during the review of this chapter. As such, RG&E would like to retain the Completion Time as proposed.*

iii. The bases were revised as follows (these are ITS Category (iv) changes):

a. Plant-specific design considerations were added including using Ginna Station nomenclature and providing consistency with the dose analyses.

b. Various wording changes were made to improve the readability and understanding of the bases.

[ITS93.iii.a and b]:

3.7Q201

What are the specific 10CFR100 limits as established by the NRC staff for Ginna per the licensing documents?

Status: [] Open

Response: *The 10CFR100 limits for Ginna are the same as for every other plant (i.e., Ginna has no exemption with respect to 10CFR 100). The only difference is with respect to the definition of "well within*

10CFR100 limits" which has been accepted for Ginna for FHA as 96 rem or approximately 33% of the allowable limit.

3.7Q202 The Completion Times for Action A.1 and A.2 are under discussion.
Status: [] Open
Response: See response to 3.7Q200.

- iv. Incorporation of approved Traveller WOG-24, C.5 Acceptable, if this is just removing the brackets from "31".

Section 3.7 Current TS

9. Technical Specification 3.1.4

- i. TS 3.1.4.4 - This specification was revised to only require shutdown to MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 8 hours versus Cold Shutdown within 40 hours consistent with the LCO Applicability. This is a Ginna TS Category (v.c) change.

[CTS9.i-?]:

3.7Q203 Where is this item in the CTS markup?
Status: [] Open
Response: Item "9.i" relates to ITS LCO 3.4.16 and not Chapter 3.7 (see Attachment B, section 3.4). CTS 3.1.4.4 is not being changed with respect to Chapter 3.7 requirements.

3.7Q204 If this is a v.c change (technical equivalence) then what is revised?
Status: [] Open
Response: See response to 3.7Q203.

3.7Q205 The MODE names for Ginna are proposed to only be changed and not the MODE numbering; so, the shutdown to MODE 3 in ITS LCO 3.7.14 is the same as existing TS 3.1.4.4.
Status: [] Open
Response: See response to 3.7Q203.

3.7Q206 The issues/changes to the existing TS pertaining to Completion Times and Applicability are covered in CTS9.ii below.
Status: [] Open
Response: See response to 3.7Q207 through 3.7Q209.

- ii. TS 3.1.4.1.c - The limit on secondary coolant activity is now required to be met in MODES 1, 2, 3, and 4 and not just when the reactor is critical or RCS temperature is $> 500^{\circ}\text{F}$. The secondary coolant activity limit is based on a steam line break and the resulting dose consequences. A RCS temperature of $> 500^{\circ}\text{F}$ is based on preventing the MSSVs from lifting following a SGTR (i.e., a RCS temperature of $> 500^{\circ}\text{F}$ is only applicable to primary system activity limits not secondary limits). In addition, if the secondary coolant activity limits are not met, TS 3.1.4.4 requires entering cold shutdown (i.e., MODE 5) within 40 hours. Requiring the secondary coolant activity limits to be met for all of MODE 4 (i.e., RCS is $> 200^{\circ}\text{F}$) provides consistency with NUREG-1431 and the current Required

Actions if the limit is exceeded. This is a Ginna TS Category (iv.a) change.

[CTS9.ii-M1]:

3.7Q207 It is acceptable to make Applicability MODES 1, 2, 3 and 4.

Status: [] Closed

Response: N/A

CTS9.ii-M2]:

3.7Q208 As noted in CTS9.i above, the changes to the existing TS are not clear. There is assumed to be only one change which is a longer Applicability.

Status: [] Open

Response: See response to 3.7Q203.

3.7Q209 It is recommended that Ginna standardize its orderly shutdown times for all LCOs as explained in ITS93.iii to LCO 3.7.17.

Status: [] Open

Response: See response to 3.7Q200.

13. Technical Specification 3.3

xvi. TS 3.3.3.1 - This was revised to only require one of the two CCW heat exchangers to be OPERABLE and to specify that the CCW loop header must also be OPERABLE. As discussed in Section C, item 82.i above, the CCW heat exchangers are 100% redundant and are separated from the CCW pump trains by a section of common piping. The CCW heat exchangers are passive devices such that any failure of a heat exchanger is bounded by a failure of the CCW piping in the loop header. The loop header is defined as the section of piping from the discharge of the pumps to the first isolation valve of each supplied component. The loop header then continues from the last isolation valve on the discharge of the supplied component to the suction of the pumps. Since there is no single active failure which must be considered for the heat exchangers, they are considered part of the CCW loop header and only one heat exchanger must be OPERABLE. Requiring the CCW loop header to be OPERABLE provides a clear and concise LCO requirement for operators. These are Ginna TS Category (v.b.13) and (v.a) changes.

[CTS13.xvii-L1]:

3.7Q210 This change is on hold pending resolution of the issues raised in ITS82.ii and ITS82.iv.

Status: [] Open

Response: See responses to 3.7Q93 through 3.7Q95 and 3.7Q99 through 3.7Q107.

3.7Q211 The deletion of existing TS 3.3.3.2.b is not specifically addressed in the CTS markup or justification. Please provide.

Status: [] Open

Response: The deletion of CTS 3.3.3.2.b is discussed in change 13.xvi above. Comment #114 has been opened to add a reference to "13.xvi" in the left margin for CTS 3.3.3.2.b.

[CTS13.xvii-M1]:

3.7Q212 The v.a change part of this justification is not clear. Is it the

definition of the common loop header?
Status:[] Open
Response: Correct, the "v.a" change relates to the CCW loop header while the "v.b.13" change relates the change in requirement for the CCW heat exchangers.

3.7Q213 If so, what has changed other than reflecting the current Ginna hardware design in the ITS for an equivalent reformatting? If not, please explain?

Status:[] Open
Response: CTS 3.3.3.1 only address the CCW pumps, heat exchangers, and "all valves, interlocks, and piping associated with the above components which are required to function during accident conditions." Since "the above components" only relates to the CCW pumps and heat exchangers, requiring the CCW loop to be OPERABLE (which includes additional piping and valves unrelated to these components) is a more restrictive requirement.

xvii. TS 3.3.3.2 - This was revised to allow 72 hours (versus 24 hours) to restore an inoperable CCW pump before requiring a plant shutdown. However, the plant is no longer allowed to remain at Hot Shutdown for 48 hours before requiring additional cooldown to Cold Shutdown conditions. As such, the total time in which a CCW pump can remain inoperable remains the same (i.e., 72 hours) but the plant is not required to begin cooldown activities after 24 hours. The only safety related functions supported by the CCW System are with respect to the RHR, SI, and CS Systems, which all allow 72 hours to restore an inoperable train. Therefore, this change provides consistency within the new specifications. This is a Ginna TS Category (v.c) change.

[CTS13.xvii-L or M?]:

3.7Q214 It is not clear whether this is less restrictive, more restrictive or just equivalent. It is so closely coupled to new Condition C that it will have to be discussed as an apparent relaxation. This appears acceptable pending resolution of the issues raised in ITS82.ii.

Status:[] Open

Response: See responses to 3.7Q93 through 3.7Q95.

xviii. TS 3.3.4.1 - This was revised to require that the six sets of motor operated isolation valves used in the SW System to be OPERABLE for the SW System to be considered OPERABLE. Credit is taken for these valves to isolate the nonessential and nonsafety related components within the SW System following a coincident safety injection and undervoltage signal. This is a conservative change which provides a clarification to licensed personnel. This is a Ginna TS Category (v.a) change.

[CTS13.xviii-M1]:

3.7Q215 The six sets of insulation valves are not readily identifiable from the line drawing of the SW System dated 12-6-91. Only four sets are found here for non-safety related components and two sets are for safety-related components. Explain further.

Status:[] Open

Response: The attached sketch has the six sets of isolation valves circled. These sets include both valves for non-safety related and safety related loads. The two sets of isolation valves for the safety related components (i.e., CCW heat exchangers and SAFW) are also required to isolate on a coincident SI and UV signal since these components are not required until at least 10 minutes following an accident. See Insert 3.7.8.9 to ITS LCO 3.7.8 bases for additional information (last paragraph on first page of Insert beginning with "The SW trains and loop header...")..

3.7Q216 Are these automatic valves which isolate on the SI or undervoltage signal?

Status:[] Open

Response: All six sets of isolation valves isolate on a coincident SI and UV signal. See Insert 3.7.8.3 to ITS LCO 3.7.8 bases.

3.7Q217 Which SR covers these valves and why not have their own separate SR?

Status:[] Open

Response: ITS SR 3.7.8.2 addresses these isolation valves since these are the only automatic valves in the SW flow paths.

3.7Q218 Need terminology protocol to clarify these valves separate from the cross-connect valves of other systems.

Status:[] Open

Response: The terminology used with respect to the SW valves is consistent with Ginna nomenclature. The cross-connect valves are manual valves as shown on the sketch.

3.7Q219 As noted in ITS83.iv.a, the background text is illegible and needs rewriting.

Status:[] Open

Response: See response to 3.7Q119.

xix. TS 3.3.4.2 - This was revised to allow one SW train comprised of two pumps and six motor operated valves supplied by the same electrical train to be inoperable for 72 hours before requiring a plant shutdown. Since the SW trains are 100% redundant, removing one of two trains only affects redundancy and does not place the plant outside the accident analyses. Since most other safety functions allow 72 hours for one train to be inoperable (e.g., ECCS trains), this change provides consistency within the new specifications. In addition, this specification was revised to address the scenario if all SW pumps or the SW loop header are inoperable. In this condition, immediate action must be initiated to restore one SW pump or the loop header to OPERABLE status; however, it is not prudent to exit the MODE of Applicability since the SW System is required in MODE 5 for decay heat removal. Instead, Required Actions have been provided to require a cooldown to MODE 4. In this lower MODE, AFW is providing for decay heat removal. If AFW were lost, additional time is required before RHR (and consequently SW) would be required. This change is also consistent with the Required Actions for loss of CCW. These are Ginna TS Category (v.c) changes.

[CTS13.xix-L1]:

3.7Q220 This is not a technically equivalent v.c change but a relaxation for

the LCO 3.7.8 Condition A which has not been justified as a less restrictive v.b change. Pleased provide as such.

Status:[]

Open

Response: *RG&E agrees that this is a less restrictive "v.b" change. Comment #115 has been opened to revise Attachment A to change the Ginna TS Category to (v.b). The justification provided in "xix" above will not change except for the last sentence.*

3.7Q221 This appears acceptable pending the resolution of the issues in ITS83.ii.

Status:[]

Open

Response: *See responses to 3.7Q109 through 3.7Q114.*

3.7Q222 Also, the CTS markup appears to be missing the new proposed Condition C.

Status:[]

Open

Response: *The markup does not specifically show new proposed Condition C but change 13.xix is provided in the left margin which specifically discusses this issue.*

xx. TS 3.3.5.1 - This was revised to require the control room emergency air treatment system (CREATS) to be OPERABLE in MODES 1 through 6 and during movement of irradiated fuel assemblies instead of only when RCS is $\geq 350^{\circ}\text{F}$. Current Ginna Station TS 3.5.6 requires that the control room HVAC detection system (i.e., chlorine, ammonia, and radioactivity monitors) be OPERABLE at all times. However, the filtration system is only required to be OPERABLE above 350°F . The filtration system is designed to ensure that dose rates to operators are within the guidelines of GDC 19 in the event of an accident. While dose rates to operators is expected to be lower when the RCS is $< 350^{\circ}\text{F}$, no current analyses exist under these conditions. In addition, failures of the waste gas decay tanks can still occur below 350°F which also require control room isolation. Therefore, the MODE of Applicability was revised to provide consistency within the specifications and the accident analyses. This is a Ginna TS Category (iv.a) change.

[CTS13.xx-M1]:

3.7Q223 It is acceptable to change the Applicability. With the removal of the brackets to MODES 5 and 6 in the proposed ITS, this is now "At all times." Why not use this phrase for clarity?

Status:[]

Open

Response: *"At all times" is not used anywhere in the ITS or NUREG since it implies Applicability when the plant is no longer in any MODE. Therefore, RG&E requests use of MODES 1-6 for this LCO.*

3.7Q224 Why is TS 3.5.6 not provided in this Chapter 3.7 submittal?

Status:[]

Open

Response: *CTS 3.5.6 relates to the actuation of the CREATS only and is addressed in ITS LCO 3.3.5.*

xxi. TS 3.3.5.2 - This was revised to provide requirements for an inoperable filtration train and inoperable dampers. The CREATS dampers isolate the control room in the event of a radiological event while the filtration train filters the control room atmosphere

following isolation. The new specification continues to allow the filtration train to be inoperable for 48 hours before requiring a shutdown or placing the control room in the emergency radiation mode (i.e., CREATS Mode 6). If one of the two redundant dampers in each outside air flow path is inoperable, the new specifications allow 7 days to restore the damper to OPERABLE status similar to restoring one train of redundant CREFS in NUREG-1431. If both dampers are inoperable, the plant must enter LCO 3.0.3 since the control room can no longer be isolated. If both dampers are lost in MODES 5 or 6, or during fuel movement, then fuel movement and CORE ALTERATIONS must be suspended immediately. These changes provide consistency with the accident analyses and NUREG-1431. These are Ginna TS Category (v.a) changes.

[CTS13.xxi-L1 or L2]:

3.7Q225 The proposed LCO for CREATS separates the portion of the filtration train from the CREAT System. This is not in the existing TS, so please explain.

Status: [] Open

Response: See response to 3.7Q134.

3.7Q226 The sketch of CREATS requested earlier is needed to complete this evaluation.

Status: [] Open

Response: See response to 3.7Q134.

14. Technical Specification 3.4

- i. TS 3.4.1 - This was revised to specifically require that all MSSVs be tested prior to entering MODE 2 versus the current wording which allows the MSSVs to be removed for testing at any time. This change is consistent with current operating practices and ensures that the MSSVs are OPERABLE before the reactor goes critical but allows the MSSVs to be tested under hot conditions (i.e., $\geq 350^{\circ}\text{F}$). In addition, the MSSV setpoints were added to the new specification since these are assumptions within the accident analyses. These are Ginna TS Category (v.a) changes.

[CTS14.i-L1]:

3.7Q227 The category v.a should be further explained because the existing TS is too ambiguous and the existing TS does not have the exact wording as is implied in the above justification. It is clear that the existing TS exempts MSSVs from being available during testing. It could be interpreted that the test is performed only when a parameter is met and that is "with the RCS temperature at or above 350°F ". The most logical interpretation is that the availability of all eight MSSVs only applies "with the RCS temperature at or above 350°F ". Based upon this last interpretation, the change requested is less restrictive because RCS temperature is no longer at or near 350°F but can be as high as 540°F just prior to entry into MODE 2.

Status: [] Open

Response: RG&E has always interpreted CTS 3.4.1 to require all eight MSSVs to be OPERABLE "with the RCS temperature at or above 350°F ." However, the MSSVs are not required to be OPERABLE during testing of the

MSSVs. ITS LCO 3.7.1 requires all eight MSSVs to be OPERABLE in MODES 1, 2, 3, and 4, but these valves are not required to be tested until prior to entering MODE 2. Even though the MSSVs are not required to be tested until prior to MODE 2, the MSSVs must still be OPERABLE to the extent practical before the test is performed above MODE 5. This is consistent with SR 3.0.1 with respect to post maintenance testing for components which cannot be tested until after the LCO MODE of Applicability has been entered. This is also consistent with the interpretation of CTS 3.4.1; otherwise, why would testing be allowed above 350°F when the LCO is required to be met as of that time.

3.7Q228 In existing TS 3.4.1, explain the modifying phrase "turbine cycle code approved steam relieving capability" added to the "eight MSSVs being available". Does this have any technical significance? Is it in lieu of just saying an "ASME code safety valve"?

Status: [] Open

Response: The "turbine cycle code" reference was contained in the original 1969 TS since "ASME code safety valve" was not in use at that time and 10 CFR 55a did not exist. However, they are intended to mean the same thing.

- ii. TS 3.4.2.1.b - This was revised to be consistent with the accident analysis assumptions as discussed in the new bases. Essentially, the accident analyses treat the preferred AFW System as four trains (i.e., two motor driven trains and two turbine driven trains) such that each SG receives flow from two AFW trains. Therefore, the failure of both motor driven trains or the turbine driven train (or both flowpaths) has the same consequence (i.e., loss of one train to each SG). Since the turbine driven train is allowed to be inoperable for up to 72 hours per TS 3.4.2.2.a (and NUREG-1431), this specification was revised to allow both motor driven AFW pumps to be inoperable for up to 72 hours. In addition, if both AFW trains to a common SG are inoperable, the new specifications allow 4 hours to restore at least one train before requiring a controlled cooldown. A time limit for being in this configuration is necessary since no AFW would be available in the event of a HELB which affects the only SG able to receive AFW. Requiring an immediate cooldown in this configuration is not considered prudent since AFW provides for decay heat removal in lower MODES. These are Ginna TS Category (v.b.14) and (v.a) changes, respectively.

[CTS14.ii-L1]:

3.7Q229 This appears acceptable for two MDAFW pumps to be inoperable for 72 hours; however, this is pending the final resolution of the agreed upon format for the LCO(s) for AFW and SAFW.

Status: [] Open

Response: See response to 3.7Q62.

[CTS14.ii-L2]:

3.7Q230 This is for the new proposed Condition E which is on hold pending the final resolution of the agreed upon format for the LCO(s) for AFW and SAFW. This is a relaxation, not a more restrictive change; so, provide justification accordingly.

Status: [] Open

Response: Condition E is not a relaxation since CTS 3.4.2.1.b allows: (1) both MDAFW and one TDAFW flowpath inoperable for 24 hours; and (2) one MDAFW and one or both TDAFW flowpaths inoperable for 24 hours. In either case, flow from the preferred AFW System is unavailable to one steam generator. Condition E limits this condition for only 4 hours.

3.7Q231 In order to enter this new Condition E, shouldn't there be a verification that both trains of SAFW are OPERABLE?

Status: [] Open

Response: This verification is not required by CTS 3.4.2.1.b. In addition, the only method of performing a verification is to perform an actual pump test which is probably not possible within a 4 hour period.

- iii. TS 3.4.2.3 - This was revised to require that the SAFW cross-tie be available when the SAFW System is required to be OPERABLE. This change is required since the accident analyses credit the use of the cross-tie for HELBs with a failure of one SAFW pump. Each cross-tie motor operated valve is considered part of the SAFW train which shares the same electrical power source. This is a Ginna Station TS Category (v.a) change.

[CTS14.iii-L1]:

3.7Q232 This appears to be a less restrictive change. Provide the identification of the set of cross-tie valves to which this change applies.

Status: [] Open

Response: By definition, expanding a current TS to include additional equipment (i.e., valves) is a less restrictive change. These cross-tie valves are credited in the accident analyses and are required to be tested in CTS 4.8.5. The attached sketch shows their locations.

3.7Q233 A unique SR will be required to verify periodically these specific valves are assured to be opened.

Status: [] Open

Response: ITS SR 3.7.5.4 specifically addresses these cross-tie valves.

3.7Q234 Are there any pressure isolation concerns if these valves remain open in the SAFW during all expected operation modes?

Status: [] Open

Response: These cross-tie valves are normally closed to maintain train independence. They are also maintained closed to prevent a passive failure in either SAFW train from failing both trains.

3.7Q235 Are the electrical and mechanical train divisions uniform without any of the SW System pumps combinations to be concerned about.

Status: [] Open

Response: The SAFW System was designed and installed in accordance with mid-1970 requirements for separation, etc. Each SAFW pump train is fully redundant and does not rely on the other pump train for any accident analysis assumption.

- iv. TS 3.4.3 - The requirement for SW suction for the AFW and SAFW pumps were relocated to the LCO for these pumps. The CSTs provide the preferred source of condensate to the preferred AFW pumps while the

SW System is the safety related source for both the preferred and standby AFW systems. The relocation of the need for a SW supply to the AFW pumps within technical specifications does not reduce the requirement. Instead, the change provides consistency within the new specifications and is easier for licensed personnel to understand. This is a Ginna TS Category (i) change.

[CTS14.iv-RI1]:

3.7Q236 Clarify what was relocated to where? For SAFW, is this Action b of TS 3.4.3 being transferred to where in LCO 3.7.5? It is not clear what is stated above for the relocation for AFW. Please explain.

Status: [] Open

Response: *CTS 3.4.3.a1 requires the CSTs to be OPERABLE. However, the CSTs do not, and cannot, supply the SAFW System as discussed in the response to 3.7Q76. Therefore, the CSTs only supply the preferred AFW System and this requirement has been relocated to ITS LCO 3.7.6. CTS 3.4.3.a2 requires SW for the SAFW System. This requirement has been relocated to the LCO bases for ITS LCO 3.7.5. Please note that SW is also required for the preferred AFW System as its safety related source and this is now a requirement for this system's operability per the bases for ITS LCO 3.7.5.*

- v. TS 3.4.3 - This was revised to require that a backup source of condensate be verified within 4 hours when the CSTs are inoperable versus demonstrating the operability of the SW System. Specifying a time limit for verifying the backup condensate source is a conservative change which now provides a clear and concise requirement for plant operators. Revising the Actions to allow any alternate source to be used as a backup source provides additional operational flexibility since other condensate sources than the SW System can be used if necessary. These sources are described in the bases for new LCO 3.7.6. These changes are consistent with NUREG-1431 and are Ginna TS Category (v.a) changes.

[CTS14.v-L1]:

3.7Q237 This is a relaxation to provide an alternate course of action before declaring a component inoperable. Provide justification as such.

Status: [] Open

Response: *CTS 3.4.3 requires demonstration of the OPERABILITY of the SW supply to the preferred AFW System within 7 days if the CSTs are unavailable. ITS LCO 3.7.6 requires verification of a backup water supply within 4 hours which is a shorter Completion Time but more flexible with respect to sources. However, the bases for ITS LCO 3.7.5 require that the preferred AFW pumps be capable of taking suction from SW within 10 minutes (see Insert 3.7.5.8) for OPERABILITY. Therefore, if the SW Supply were unavailable to the preferred AFW pumps, they would be declared inoperable regardless of the status of the CSTs. ITS LCO 3.7.6 Required Action A.1 can thus be met by using an additional source of water beyond the OPERABLE SW supply or by opening the SW supply itself.*

3.7Q238 The questions on ITS81.ii need addressing and ITS81.iv needs the reverification proposed to be deleted and not justified above.

Status: [] Open

Response: *See response to 3.7Q76 through 3.7Q78 and 3.7Q81.*

21. Technical Specification 3.11

- i. TS 3.11.1 - This was revised to require that the Auxiliary Building Ventilation System (ABVS) be OPERABLE when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. The specific components which are required for the ABVS to be considered OPERABLE were relocated to the bases similar with the structure of NUREG-1431 and the ITS Writer's Guide. The bases for LCO 3.7.10 now require that one of the two 100% capacity Auxiliary Building main exhaust fans, exhaust fan C, the SFP Charcoal Absorber System, and all associated ductwork, valves and dampers be OPERABLE. In addition, TS 3.11.1.c was revised to require a negative pressure within the Auxiliary Building operating floor with respect to the outside environment instead of requiring all doors, windows, and other direct openings between the operating floor area and the outside to be closed. This change provides consistency with assumptions of the fuel handling accident as described in the bases: This change also provides a much clearer specification which is easier for licensed personnel to read and understand without any reduction in actual requirements. These are Ginna TS Category (i) and (v.a) changes, respectively.

[CTS21.i-RI1]:

3.7Q239 It appears that after irradiated fuel has decayed more than 60 days, ABVS and all components are turned off. This was not the assumption of the improved TS per NUREG-1431. If irradiated fuel is handled after 60 days and a FHA occurs, explain the need for the LCO.

Status: [] Open

Response: See responses to 3.7Q156, 3.7Q157, and 3.7160. I would also suggest the reviewer look at Inserts 3.7.10.3 and 3.7.10.4 and withdraw his last statement.

3.7Q240 It is acceptable to relocate OPERABILITY requirements to the BASES.

Status: [] Closed

Response: N/A

3.7Q241 The operability requirements of existing TS 3.11.1.a thru e will be verified after the sketches of the system are received as requested in ITS88.ii, Item 1.

Status: [] Open

Response: Sketch has been provided in response to 3.7Q155.

[CTS21.i-M1]:

3.7Q242 It is acceptable to require a negative pressure verification in lieu of verifying doors, dampers, and etc, are closed. Where is this placed in the LCO? As noted earlier in ITS88, a Surveillance Requirement(s) is needed.

Status: [] Open

Response: The negative pressure requirement is located in item c of the LCO bases for ITS LCO 3.7.10. This is verified by ITS SR 3.7.10.1 (see bases and attached procedure RF-8.4, step 3.10).

- ii. TS 3.11.2 - The requirement to continuously monitor radiation levels in the SFP area was not added to the new specifications. No

screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the SFP radiation levels only provide a backup source to a SFP problem. Other LCOs provide adequate verification of SFP primary indications (i.e., level and boron concentration) which ensure that all accident analysis assumptions are met. Since a fuel handling accident can only occur as a result of fuel movement, personnel would be stationed within the Auxiliary Building and immediately aware of a problem. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.

[CTS21.ii-R01]:

3.7Q243 Which radiation monitors are involved here? Are they on the plant vent, main auxiliary building exhaust fans or local area monitors?

Status: [] Open

Response: *The monitor is a constant air monitor for particulate, gas and iodine (see attached procedure RF-8.4, step 3.14). This is a portable monitor only.*

- iii. TS 3.11.3 and 3.11.5 - The heavy load restriction for movement of loads over the SFP was not added to the new specifications. No screening criteria apply for this requirement because the heavy load limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This change is consistent with WCAP-11618 (Ref. 52) and is a Ginna TS Category (iii) change.

[CTS21.iii-R02]:

3.7Q244 This appears acceptable; but, what is the topic of the WCAP-11618 and the consistency noted here.

Status: [] Open

Response: *WCAP-11618 is the application of the criteria to the Westinghouse Standard TS (i.e., NUREG-0452, Revision 4). This WCAP was used by the industry and NRC to create NUREG-1431, Revision 0.*

- iv. TS 3.11.4 - The SFP water temperature limit was not added to the new specifications. No screening criteria apply for this requirement because the SFP water temperature limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.

[CTS21.iv-R01]:

3.7Q245 Please provide a copy of Ginna's screening criteria for this existing TS requirement.

Status: [] Open

Response: *The SFP water temperature limit is not installed instrumentation that is used with respect to degradation of the reactor coolant*

pressure boundary. This limit is also not an initial condition of a DBA or part of the primary success path to mitigate a DBA. The only DBA considered with respect to the SFP is a fuel handling accident. The fuel handling accident only considers temperature with respect to moderator density such that low temperatures are conservatively used in the analysis. If boiling of the SFP were to occur, criticality is prevented by the storage array in place and the design of the fuel. The SFP temperature also has not been shown by operating experience or probabilistic safety assessment to be significant to public health and safety. Therefore, the SFP water temperature limit does not meet the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM.

3.7Q246 ITS92.ii, item 3, is related to this existing TS requirement. There appears to be an analysis which assumes the initial SFP temperature to be first 120°F and then 150°F because at 180°F the structural integrity of the fuel pool is not maintained as noted in UFSAR Section 9.1.3.1.

Status: [] Open

Response: See response to 3.7Q194.

28. Technical Specification 4.1

ii. The following changes were made to TS 4.1.2 or Table 4.1-2:

a. Table 4.1-2, #6a was revised to extend the

g. Table 4.1-2, Functional Unit #17 was revised to only require verification of SFP boron concentration once every 31 days when fuel is stored in the SFP and the position of fuel assemblies which were moved in the SFP have not been verified. The current monthly requirement (regardless of the status of the SFP verification) is not reflected in the fuel handling accident analysis which does not credit the availability of soluble boron. This is a Ginna TS Category (v.b.32) change.

[CTS28.ii.g-L1]:

3.7Q247 There appears to be confusion over what is the boron concentration limit for the SFP. UFSAR Section 9.1.2.2.1 states SFP is maintained at least at a 2000 ppm concentration. Appendix F in the proposed COLR says equal to or greater than 300 ppm. The above justification assumes no soluble boron as does existing TS 5.4.2. Existing TS 5.4.6 says SFP concentration matches that in the reactor cavity. With so many disagreements, how can this be placed in the COLR?

Status: [] Open

Response: See response to 3.7Q183 with respect to difference between the UFSAR and Appendix F of the submittal. The SFP concentration is only required to match the reactor cavity boron concentration when the reactor vessel head is removed and fuel movement is being performed between the SFP and the reactor vessel. Otherwise, the SFP is isolated from the reactor cavity by use of a blind flange and gate valve.

i. The following new requirements were added to Table 4.1-2 (Ginna TS Category (iv.a) changes):

1.

12. SR 3.7.11.1 - requires verification every 31 days that ≥ 23 feet of water is available above the top of the irradiated fuel assemblies seated in the storage racks during fuel movement in the SFP. This verification is required since the fuel handling accident assumes that at least 23 feet of water is available with respect to iodine releases.

[CTS28.ii.i.17-M1]:

3.7Q248 It is acceptable to add this SR pending resolution of issues identified in ITS90.iii.

Status: [] Open

Response: See response to 3.7Q181.

13. SR 3.7.13.1 and SR 3.7.13.2 - verification prior to fuel movement in the SFP that the associated fuel assembly meets the necessary requirements for storage in the intended region (e.g, enrichment limit, burnable poisons present). This verification is required to limit the amount of time that a fuel assembly could be misloaded in the SFP.

[CTS28.ii.i.13-M1]:

3.7Q249 It is acceptable to add these SRs pending resolution of issues identified in ITS92.ii.

Status: [] Open

Response: See responses to 3.7Q192 through 3.7Q195.

14. SR 3.7.6.1 - requires verification every 12 hours that the CST volume is $\geq 22,500$ gallons. This ensures that the minimum volume of condensate is available for the preferred AFW System following an accident.

[CTS28.ii.i.14-M1]:

3.7Q250 It is acceptable to add this SR pending resolution of issues identified in ITS81.ii.

Status: [] Open

Response: See responses to 3.7Q76 through 3.7Q78.

15. SR 3.7.7.1 - requires verification every 31 days that each CCW manual and power operated valve in the CCW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the CCW System is capable of performing its function following a DBA to provide cooling water to safety related components.

[CTS28.ii.i.15-M1]:

3.7Q251 It is acceptable to add this SR pending resolution of issues identified in ITS82.iii.

Status:[] Open

Response: See responses to 3.7Q96 through 3.7Q98.

16. SR 3.7.7.2 - requires performance of a complete cycle of each CCW motor operated isolation valve to the RHR heat exchangers in accordance with the IST Program. This ensures that the normally closed motor operated valves are capable of being opened following a DBA.

[CTS28.ii.i.16-M1]:

3.7Q252 It is acceptable to add this SR pending resolution of issues identified in ITS82.iii.

Status:[] Open

Response: See responses to 3.7Q96 through 3.7Q98.

17. SR 3.7.8.1 - requires verification every 31 days that each SW manual and power operated valve in the SW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the SW System is capable of performing its function following a DBA to provide cooling water to safety related components.

[CTS28.ii.i.17-M1]:

3.7Q253 It is acceptable to add this SR pending resolution of issues identified in ITS83.v.

Status:[] Open

Response: See response to 3.7Q119 through 3.7Q125.

28.ii.j.

- m. Table 4.1-2, Functional Unit #12 - This was relocated to the TRM since it does not meet any of the requirements for inclusion in the ITS. This is a Ginna TS Category (iii) change.

[CTS28.ii.m-R01]:

3.7Q254 It is agreed this functional unit does not meet Tech Spec criteria for inclusion; however, has Ginna complied with the requirements of Generic Letter 86-10? Has the Fire Protection program been reviewed and approved by the NRC staff? After acceptance then the elements of the fire Protection Program can be relocated.

Status:[] Open

Response: The fire protection requirements have all been relocated from the TS by Amendment No. 49 (see Ginna license page 3).

- n. Table 4.1-2, Functional Unit #8 - The Frequency for determining gross specific activity of the secondary system was revised from once every 72 hours to once every 31 days. In addition, the determination of I-131 was also changed to once every 31

days independent of the last activity level since the current Ginna TS allow up to 6 months between tests. These changes are all consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.

[CTS28.ii.n-L1]:

3.7Q255 The proposed SR for a specific isotopic analysis every 31 days is acceptable and this is a technically equivalent to the note (3) of the table.

Status: [] Closed

Response: N/A

3.7Q256 The SR for determining gross specific activity is not in Proposed ITS LCO 3.7.14. The elimination of the more frequent gross activity test every 72 hours has not been justified as a less restrictive change to the existing TS. Please provide this justification.

Status: [] Open

Response: *RG&E proposes to revise item n above as follows:*

Table 4.1-2, Functional Unit #18 - The determination of I-131 was changed to once every 31 days independent of the last activity level since the current Ginna TS allow up to 6 months between tests. This is a Ginna TS Category (v.c) change consistent with NUREG-1431. This surveillance was also revised to relocate the requirement to perform gross activity tests of the secondary coolant once every 72 hours to Ginna procedure CH-PRI-SCHED (attached). This surveillance is mainly in the CTS to provide early indication of changes to I-131 equivalent activity which is what is evaluated in the accident analyses. I-131 equivalent activity is determined every 6 months in the CTS unless the gross activity test indicates that I-131 equivalent activity is > 10% of its allowable limit at which time the surveillance changes to monthly. Since I-131 equivalent activity is now to be tested monthly regardless of its percentage with respect to allowable limits, this early indication of changes as provided by the gross activity test is no longer required. This is a Ginna TS Category (iii) change.

32. Technical Specification 4.5

xii. TS 4.5.2.3.9 - This was revised to require a test of the automatic actuation capability of the CREATS once every 24 months. This verification is necessary to ensure that the control room environment can be isolated in the event of a radiological release. This is a Ginna TS Category (iv.a) change.

[CTS32.xii-M1]

3.7Q257 Please verify that the existing TS 4.5.2.3.9 was retained as is and this should state that "A new SR was added to test the automatic actuation capability of CREATS at refueling."

Status: [] Open

Response: *RG&E proposes to revise item xii above as follows:*

TS 4.5.2.3.9 - This was revised to require a test of the automatic actuation capability of CREATS once every 24 months instead of monthly. Verification of this instrumentation on a refueling outage basis instead of monthly is evaluated in Attachment H. This is a

Ginna TS Category (v.b) change. The requirement to operate the system at least 15 months every month has been retained (ITS SR 3.7.9.1).

34. Technical Specification 4.7

- i. TS 4.7 was revised to include a surveillance to ensure that each MSIV can close on an actual or simulated actuation signal every 24 months consistent with NUREG-1431 and current Ginna Station TS Table 3.5-2 which require that the isolation signals to the MSIVs be OPERABLE. In addition, Required Actions were provided in the event that the MSIVs cannot close as required by this Surveillance. These actions require restoration of, or closure of an inoperable MSIV, within 24 hours. In the event that both MSIVs are inoperable, the plant must enter LCO 3.0.3. Finally, requirements for the main steam non-return check valves were added. These are Ginna TS Category (iv.a) changes. The test to ensure that each MSIV can close on an actual or simulated actuation signal every refueling outage is judged as technically equivalent and not more restrictive.

[CTS34.i-M1]:

3.7Q258

The existing TS 4.7 appears to have been clarified rather than revised in a more restrictive manner. Since no LCO existed, Applicability was added. Proposed SR 3.7.2.3 is the same as "The MSIVs shall be tested at least each refueling outage".

Status: []

Open

Response:

Providing Applicability where none previously existed is a more restrictive change. In addition, the CTS bases state that the closure time for this surveillance is "consistent with expected response time for instrumentation" and does not require this test to be conducted by use of an isolation signal. Due to the design of these valves, this test could be done by removing DC control power to their respective solenoid valves and ensuring that the valve closes within 5 seconds and still meet this surveillance. However, current Ginna practice is to perform the test by use of the manual controls in the control room.

[CTS34.i-L1]:

3.7Q259

The existing TS 4.7, Main Steam Isolation Valves (MSIVs) covers only testing requirements and does not have a corresponding Limiting Condition of Operation for MSIVs. Table 3.5-2 required MSIV operability "as open above 350°F T_{avg}". The improved TS 3.7.2 was added to limit the operability of the MSIVs to just MODES 1, 2, and 3. New Conditions with less restrictive Required Actions were added to preclude an immediate shutdown if a MSIV became inoperable. Provide justifications for these relaxations.

Status: []

Open

Response:

CTS Table 3.5-2 relates to the instrumentation to the MSIVs only, not to the actual valves (i.e., the instrumentation to the valve could be OPERABLE while the valve could not be able to physically close). Therefore, the CTS do not currently have any MSIV requirements except for instrumentation and the surveillance discussed above.

[CTS34.i-L2]:

3.7Q260

See questions concerning the addition of these new valves in ITS77.viii. The existing TS 4.7 was changed to add new Limiting Conditions of Operation for the "Non-Return Check Valves". The improved TS 3.7.2 was further modified to add new Conditions with less restrictive required actions to preclude an immediate shutdown if the main steam header is nonisolable with the Non-Return Check Valves becoming inoperable. Provide justifications for these relaxations.

Status: [] Open

Response: *See responses to 3.7Q33 through 3.7Q36. Also, since the CTS have no requirements for OPERABILITY of these non-return check valves, there is no relaxation with respect to the CTS.*

35. Technical Specification 4.8

- i. TS 4.8.1 and 4.8.2 - The Frequency of the AFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.

[CTS35.i-L1 and L2]:

3.7Q261

This justification only deals with the relocation of test requirements. The relaxation for the test intervals in TS 4.8.1 and 4.8.2 have never been justified.

Status: [] Open

Response: *The IST program currently requires quarterly tests on the AFW and SAFW pumps and valves such that this is actually a less restrictive change following implementation. The justification for this change is that ASME testing requirements only specify quarterly tests of pumps and valves as being adequate to demonstrate continued component OPERABILITY. The NRC has generically approved these testing frequencies via 10 CFR 50.55a and approval of the Ginna Station IST Program. See also change D.80.x.*

3.7Q262 This is viewed as a major relaxation to the test frequency of the motor-driven and turbine-driven AFW pumps.

Status: [] Open

Response: *See response to 3.7Q261.*

3.7Q263 The issues of relocation will not be dealt with until this relaxation is first acceptable by the NRC technical staff.

Status: [] Open

Response: *See response to 3.7Q261.*

- ii. TS 4.8.3 - This Surveillance was revised to relocate the Frequency of testing the AFW suction and discharge valves to the Inservice Testing Program which provides sufficient control of these testing activities. In addition, the cross-over motor operated isolation valves were not added to the new specifications since these valves

are not credited in the accident analyses (see bases for new LCO 3.7.5). These are Ginna TS Category (iii) and (v.b.39) changes, respectively.

[CTS35.ii-L1]:

3.7Q264 There is no justification for changing the test intervals from monthly per TS 4.8.3 to in accordance with the IST Program. Please provide.

Status: [] Open

Response: See response to 3.7Q261.

3.7Q265 In ITS80.i and ITS80.iii, issues pertaining to this change need to be resolved concurrently.

Status: [] Open

Response: See responses to 3.7Q62 through 3.7Q64 and 3.7Q66 through 3.7Q68.

3.7Q266 The issues of relocation will not be dealt with until this relaxation is first accepted.

Status: [] Open

Response: See response to 3.7Q261.

- iii. TS 4.8.4 - The Frequency of the SAFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.

[CTS35.iii-L1]:

3.7Q267 This justification only deals with the relocation of test requirements. The relaxation for the test interval in TS 4.8.4 has never been justified.

Status: [] Open

Response: See response to 3.7Q261.

3.7Q268 This is viewed as a major relaxation to the test frequency of the SAFW pumps.

Status: [] Open

Response: See response to 3.7Q261.

3.7Q269 The issues of relocation will not be dealt with until this relaxation is first acceptable by the NRC technical staff.

Status: [] Open

Response: See response to 3.7Q261.

- iv. TS 4.8.5 - This Surveillance was revised to relocate the Frequency of testing the SAFW suction, discharge, and cross-over valves to the Inservice Testing Program which provides sufficient control of these testing activities consistent with NUREG-1431. This is a Ginna TS Category (iii) change.

[CTS35.iv-L1]:

3.7Q270 There is no justification for changing the test intervals from monthly per TS 4.8.5 to in accordance with the IST Program. Please provide.

Status:[] Open

Response: See response to 3.7Q261.

3.7Q271 In ITS80.i and ITS80.iii, issues pertaining to this change need to be resolved concurrently.

Status:[] Open

Response: See responses to 3.7Q62 through 3.7Q64 and 3.7Q66 through 3.7Q68.

3.7Q272 The issues of relocation will not be dealt with until this relaxation is first accepted.

Status:[] Open

Response: See response to 3.7Q261.

- v. TS 4.8.6 - This was revised to relocate the acceptance criteria for the AFW and SAFW tests to the actual procedures performing these tests. The new bases identify what is required for OPERABILITY of the AFW and SAFW Systems such that specifying this acceptance criteria is unnecessary. In addition, both the bases and test procedures are controlled under 10 CFR 50.59. This is a Ginna TS Category (iii) change.

[CTS35.v-RI1]:

3.7Q273 It is acceptable to relocate the details of these testing procedures to the BASES. Please identify where this is located.

Status:[] Open

Response: The acceptance criteria for these tests have been relocated to the following procedures: PT-16Q-A, PT-16Q-B, PT-16Q-T, PT-36Q-C, and PT-36Q-D. The applicable pages of these procedures are attached.

3.7Q274 The deleted text in the last sentence is not acceptable with this justification. This on hold pending the resolution of the issues in CTS35.i and CTS35.iii.

Status:[] Open

Response: See response to 3.7Q261.

- vi. TS 4.8 - A new Surveillance was added requiring verification every 31 days of the correct position of each AFW and SAFW manual, power operated and automatic valve in the flow path that is not locked, sealed or otherwise secured in position. This verification is required to ensure that the AFW and SAFW Systems are OPERABLE when not in service. This is a Ginna TS Category (iv.a) change.

[CTS35.vi-M1]:

3.7Q275 It is acceptable to add this new SR 3.7.5.1.

Status:[] Closed

Response: N/A

38. Technical Specification 4.11

- iii. TS 4.11.1.1.d - This was not added to the new specifications since this verification is not required to ensure that initial assumptions of the accident analyses are still met. The SFP Charcoal Absorber

System does not utilize heaters. The bases for SR 3.7.13.1 state that operating the ventilation system for ≥ 15 minutes every 31 days for systems without heaters is to ensure system operation. In accordance with new LCO 3.7.10 (NUREG-1431 LCO 3.7.13), the ABVS is required to be in operation during fuel movement within the Auxiliary Building. As such, the ABVS is not a standby system at Ginna Station (i.e., the system must be both OPERABLE and in operation during its MODE of Applicability). Therefore, a monthly verification provides no verification of any accident analysis assumption. Instead, a new Surveillance was added which requires verification every 24 hours that the Auxiliary Building operating floor level is at a negative pressure with respect to the outside environment. This verification is consistent with plant practices and ensures that an initial assumption of the fuel handling accident is being maintained. The change is also consistent with Reference 55. This is a Ginna TS Category (v.c) change.

[CTS38.iii-L1]:

3.7Q276 Explain the filter bank justification. Performing the verification provides assurance the filter bank is operational if called upon to operate. While the ABVS is being operated, this SFP Charcoal Adsorber System appears capable of being put into a bypass or standby and never checked if it were not for existing TS 4.11.1.d.

Status: [] Open

Response: See responses to 3.7Q154 and 3.7Q156.

3.7Q277 Why are the Auxiliary Building Charcoal Filters not tested also?

Status: [] Open

Response: See responses to 3.7Q154 and 3.7Q156.

66. New Requirements (Ginna TS Category (iv.a) Changes)

- ii. LCO 3.7.3 and the associated surveillances were added for the MFW pump discharge valves (MFPDVs), MFW regulating valves, and the associated bypass valves. This new requirement specifies an isolation time of 80 seconds for the MFPDVs and 10 seconds for the remaining valves and requires them to be OPERABLE above MODE 4 to provide isolation capability as assumed in the accident analyses.

[CTS66.ii-M1]:

3.7Q278 It is acceptable to add this LCO pending resolution of the issues raised in ITS78.

Status: [] Open

Response: See responses to 3.7Q39 through 3.7Q51.

- iii. LCO 3.7.4 and the associated surveillance were added for the atmospheric relief valves (ARVs). The LCO requires that the ARVs be OPERABLE when RCS average temperature is $> 500^{\circ}\text{F}$ in MODE 3 to provide cooldown capability following a SGTR event as assumed in the accident analyses. A Surveillance to verify that each ARV is capable of opening and closing once every 24 months was also added.

[CTS66.iii-M1]:

3.7Q279 It is acceptable to add this LCO pending resolution of the issues raised in ITS79.

Status: [] Open
Response: See responses 3.7Q52 through 3.7Q61.

iv. A COLR was developed which contains the actual limits for LCOs associated with reactor physic parameters that may change with each refueling. To prevent the need to revise Technical Specifications for parameters which are calculated using NRC approved methodology, Generic Letter 88-16 (Ref. 56) allows these limits to be relocated from the technical specifications. A copy of the proposed Ginna Station COLR is provided in Attachment F. The following parameters were relocated to the COLR:

- a. SHUTDOWN MARGIN
- b. MODERATOR TEMPERATURE COEFFICIENT
- c. Shutdown Bank Insertion Limit
- d. Control Bank Insertion Limits
- e. Heat Flux Hot Channel Factor
- f. Nuclear Enthalpy Rise Hot Channel Factor
- g. AXIAL FLUX DIFFERENCE
- h. Overtemperature ΔT and Overpower ΔT Trip Setpoints
- i. RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits
- j. Accumulator Boron Concentration
- k. RWST Boron Concentration
- l. Spent Fuel Pool Boron Concentration
- m. Refueling Boron Concentration

[CTS66.iv.1-L1]:

3.7Q280 This is a relaxation for Spent Fuel Pool boron concentration which has not been justified as such here.

Status: [] Open

Response: *The SFP boron concentration limit is not in the CTS such that there is no relaxation with respect to the CTS.*

3.7Q281 Also, the value to be in the COLR is in dispute. Please see CTS28.ii.g and ITS91.ii.

Status: [] Open

Response: See response to 3.7Q183 and 3.7Q247.

Section 3.8 TS

3.8Q1 - The CTS 3.0.2 exclusion for the OPERABILITY of a system, subsystem, train, component, or device when one of its power sources is inoperable is carried over to ITS LCO 3.8.1. The justification for changing the allowable outage time from the current 1 hour to the proposed 12 hours is its consistency with NUREG-1431. This is not enough justification for this less restrictive change. Provide justification for this change based on plant-specific design capabilities.

Response: *CTS 3.0.2 provides an exclusion for declaring a component inoperable if its offsite power source or diesel generator (DG) source is inoperable provided that two conditions are met. These conditions are that: (1) either the corresponding offsite source or DG remains OPERABLE, and (2) the redundant component remains OPERABLE with*

either its associated offsite power or DG source OPERABLE. If either of these conditions are not met for greater than 1 hour, then the plant must initiate shutdown actions. This requirement was relocated to ITS LCO 3.8.1.

Condition A of LCO 3.8.1 applies if an offsite power source is unavailable to one or more 480 V safeguards buses. Required Action A.1 states that if any features on an unaffected bus are declared inoperable at the time of, or following, the loss of offsite power, then the redundant component on the 480 V bus which has lost offsite power must be declared inoperable within 12 hours. The NUREG-1431 bases state a Completion Time of 12 hours is acceptable since it allows the operator time to evaluate and repair any discovered inoperabilities. RG&E agrees with this basis. In order to lose a safety function in this instance, either the DG to the 480 V bus which has lost offsite power must also be inoperable or the redundant component on the affected bus must fail. The probability of either event occurring in 12 hours is very low, especially coincident with an accident. However, if the DG or the redundant component were declared inoperable, then the safety function determination program would require immediate entry into LCO 3.0.3. It should be noted that the DG is the assumed source of power in the accident analyses and not offsite power (except for cases where the availability of offsite power is the worst case). Therefore, in Condition A, even though 12 hours is allowed before declaring a component inoperable due strictly to its offsite power source being lost, the safety related source of power to that component remains available.

Condition B of LCO 3.8.1 applies if a DG is unavailable. Required Action B.2 states that if any features on an unaffected bus are declared inoperable at the time of, or following, the loss of the DG, then the redundant component on the 480 V bus which has lost the DG must be declared inoperable within 4 hours. The NUREG-1431 bases state a Completion Time of 4 hours is acceptable since it allows the operator time to evaluate and repair any discovered inoperabilities. RG&E agrees with this basis. In order to lose a safety function in this instance, either the offsite power source to the 480 V bus with an inoperable DG must fail or the redundant component on the affected bus must fail. The probability of either event occurring in 4 hours is very low, especially coincident with an accident. However, if offsite power were lost or the redundant component declared inoperable, then the safety function determination program would require immediate entry into LCO 3.0.3. It should be noted that Ginna Station has two available sources of offsite power, including backfeeding through the main transformer which decreases the potential for this scenario.

Condition C of LCO 3.8.1 applies if no offsite power is available to one or more 480 V safeguards buses and one DG is declared inoperable. If offsite power and a DG were lost to the same bus, the associated components would all be declared inoperable. If separate buses were affected, both Conditions A and B would be entered which essentially allows either 4 or 12 hours in this configuration with an inoperable component. The use of 4 or 12

hours versus 1 hour is considered acceptable due to the low probability of the event occurring as discussed above.

3.8Q2 - The CTS 3.7.2.2.c time to reenergize safety-related 480-Vac buses 14, 16, 17, or 18 is 1 hour. The ITS LCO 3.8.9, Condition A, completion time is 8 hours to restore the AC electrical power distribution system to OPERABLE status. The justification given, D.17.v, states that 8 hours is consistent with the ITS. It is not clear how consistency with the ITS makes this ITS completion time acceptable. Justify the proposed completion time based on plant-specific design capabilities.

Response: The NUREG-1431 bases for LCO 3.8.9, Required Action A.1 state that an 8 hour Completion Time to restore the AC electrical power distribution system train to OPERABLE status is acceptable because:

- a. The potential for decreased safety if the plant operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

RG&E agrees with this basis since the operators should be provided with every opportunity to restore the inoperable AC electrical train before requiring a plant shutdown with only one train available. Also, during the 8 hour Completion Time, the redundant electrical train remains capable of performing its safety related function. The probability of an accident within this 8 hours is very low. In addition, it should be noted that CTS 3.7.2.2.c only applies to the 480 V safeguards buses and not to the motor control centers (MCCs) and distribution panels supplied by these buses (i.e., there is no CTS requirement for an AC electrical distribution train). Consequently, if a MCC (as supplied by any of the four 480 V safeguards buses) were declared inoperable, Ginna Station could currently enter the LCOs of the components supplied by the MCC which generally have Completion Times of 72 hours or greater. Therefore, applying ITS LCO 3.8.9 with respect to the MCCs is a more restrictive change.

3.8Q3 - The CTS 3.7.2.2.d time to achieve cold shutdown is 36 hours with both offsite sources inoperable. The ITS LCO 3.8.1, Condition D, Completion Time to attain Mode 5 (cold shutdown) is 36 hours from entering Condition D, that is, the Required Action and Completion Time of Condition A, no offsite power to one or more 480-Vac safeguards buses, is not met. Condition A, with two completion times (12 hours [if concurrent with inoperable redundant required safety features] and 72 hours to restore the offsite circuit to Operable status) results in a total time to achieve Mode 5 of 48 hours or 82 hours. This increase in the allowed time to reach cold shutdown was not justified. Discuss this change and provide justification appropriate to the change.

Response: CTS 3.7.2.2.d applies if both offsite power sources are inoperable above cold shutdown. This specification states that one offsite



power source must be restored within 72 hours. If this is not achieved, then the plant must be brought to at least hot shutdown within the next 6 hours and be in cold shutdown (i.e., MODE 5) within the following 30 hours. Therefore, the time to achieve MODE 5 is at most 108 hours from the time in which both offsite power sources were discovered inoperable (i.e., 72 hours + 6 hours + 30 hours). This is the same time as proposed in ITS. That is, Condition A requires restoration of the offsite power source within 72 hours. If this is not achieved, then Condition D requires that the plant be in MODE 3 within 6 hours and MODE 5 within 36 hours. The time to achieve MODE 5 is therefore 72 hours + 36 hours or 108 hours. Consequently, there is no difference between the CTS and ITS.

It appears that the "or" in CTS 3.7.2.2.d appears to be the source of confusion. This "or" begins at the end of the 72 hour Completion Time to restore at least one offsite power source and not from the time at which the Condition is entered (i.e., both offsite power sources are discovered inoperable). It is recognized that the ITS would consider this "or" to be in effect at the time of entering the Condition which is not how the CTS are implemented or used.

3.8Q4 - The CTS 3.7.2.2.f time to achieve cold shutdown is 36 hours with an inoperable inverter. The ITS LCO 3.8.1, Condition C, Completion Time to attain Mode 5 (cold shutdown) is 36 hours from the time Condition C is entered, that is, the Required Action and Completion Time of Condition A, one inverter inoperable, is not met. Condition A, with three completion times (2 hours, 24 hours, and 72 hours to restore the inverter) results in a total time to achieve Mode 5 of 38 hours, 60 hours, or 108 hours, respectively. This increase in the allowed time to reach cold shutdown was not justified. Discuss this change, the equivalency of the ITS Condition C Actions and Completion Times to the CTS 3.7.2.2.f ("OTHERWISE" implies a choice to immediately pursue cold shutdown) and provide justification appropriate to the change.

Response: CTS 3.7.2.2.f states that with either Instrument Bus A or C not energized from its associated inverter, the plant must re-energize the bus within 2 hours and re-energize the instrument bus from a safety-related supply within 24 hours and re-energize the bus from its associated inverter within 72 hours. The "otherwise" statement which follows these required actions means that if any of these required actions are not met, then the plant must be in hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. The "otherwise" statement does not mean to immediately pursue cold shutdown upon entering the Condition. This is always an option available for plant operators but if the "otherwise" statement were interpreted to require this shutdown path, what would be the required action if CTS 3.7.2.2.f.1, 3.7.2.2.f.2 or 3.7.2.2.f.3 were not met?

ITS LCO 3.8.7 Condition A provides the same Required Actions and Completion Times as CTS 3.7.2.2.f.1, 3.7.2.2.f.2 or 3.7.2.2.f.3 while Condition C provides the same Required Actions and Completion Times as CTS 3.8.2.2.f.4. Therefore, there is no difference between



the CTS and ITS.

- 3.8Q5 - The CTS 3.7.2.2.g time to achieve cold shutdown is 36 hours with the constant voltage transformer inoperable. The ITS LCO 3.8.1, Condition B, Completion Time to attain Mode 5 (cold shutdown) is 36 hours from the time Condition B is entered, that is, the Required Action and Completion Time of Condition A, the constant voltage transformer becomes inoperable, is not met. Condition A, with two completion times (2 hours and 7 days to restore the transformer) results in a total time to achieve Mode 5 of 38 hours or 8-½ days, respectively. This increase in the allowed time to reach cold shutdown was not justified. Discuss this change, the equivalency of the ITS Condition B Actions and Completion Times to the CTS 3.7.2.2.g ("OTHERWISE" implies a choice to immediately pursue cold shutdown) and provide justification appropriate to the change.

Response: CTS 3.7.2.2.g states that with Instrument Bus B not energized from its associated inverter, the plant must re-energize the bus within 2 hours and re-energize the instrument bus from its associated inverter within 7 days. The "otherwise" statement which follows these required actions means that if any of these required actions are not met, then the plant must be in hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. The "otherwise" statement does not mean to immediately pursue cold shutdown upon entering the Condition. This is always an option available for plant operators but if the "otherwise" statement were interpreted to require this shutdown path, what would be the required action if CTS 3.7.2.2.g.1 or 3.7.2.2.g.2 were not met?

ITS LCO 3.8.7 Condition B provides the same Required Actions and Completion Times as CTS 3.7.2.2.g.1 and 3.7.2.2.g.2 while Condition C provides the same Required Actions and Completion Times as CTS 3.8.2.2.g.3. Therefore, there is no difference between the CTS and ITS.

- 3.8Q6 - The ITS changes the CTS 18-month interval for diesel generator inspection (4.6.1.e.1), diesel generator load rejection testing (4.6.1.e.2), and diesel generator simulated loss of power with concurrent safety injection testing (4.6.1.e.3) to a frequency of 24-months in the ITS. Give justification that shows the increase in this interval will not lower diesel generator performance and its ability to meet design requirements.

Response: This response is organized into several parts. First, the DG inspection as required by CTS 4.6.1.e.1 is being relocated to the TRM (see change C.33.iv on page 240 of Attachment A). RG&E is participating in a program to develop performance based diesel generator inspection criteria instead of the current time directed inspection. This program is being developed with the full support of the DG vendor (Coltec-Fairbanks Morse / ALCO) and six other participating utilities. As such, RG&E believes that the DG inspection frequency is best controlled outside of the technical specifications relative to the actual performance of the DG. If the DG performance would require more frequent inspections than once every 24 months, RG&E would pursue the necessary actions required by

this performance based program to restore DG performance.

Second, the records related to performance of the diesel generator load rejection testing (CTS 4.6.1.e.2) have been reviewed with no failures observed since this test was first performed in 1969. While this test has historically been performed on an annual basis due to 12 month refueling cycles, RG&E has not found any historical information which would refute an increased surveillance interval of 24 months. If DG load rejection performance were to decline following the change to 24 months, the necessary actions would be implemented via the program discussed in the first section above or via implementation of the Maintenance Rule which is required by June 1996.

Third, the records for the diesel generator loss of power with concurrent SI test (CTS 4.6.1.e.3) have been reviewed for the past 11 years (i.e., 1985 - 1995). These tests are currently conducted annually with only two relay failures observed during this time frame as discussed below:

- a. In 1994, the breaker for the non-essential boric acid evaporator failed to completely trip during the test and was damaged. The associated DG successfully started and the load was subsequently relocated to a non-DG supplied bus.*
- b. In 1995, the MCC C load shed relay initially shed all non-essential loads but then began to "chatter" such that certain loads could have become reconnected. Engineering analyses demonstrated that even assuming worst case conditions, the load shed relay failure would not have prevented the associated DG from performing its safety related function.*

Therefore, all the failures observed during this 11 year period would not have prevented the DG from performing its required function. RG&E has also implemented a reliability centered maintenance program which includes trending and root cause evaluation of equipment failures. The new Maintenance Rule requires similar programs to ensure the continued reliability of the diesel generators. These programs will ensure continued DG reliability.

3.8Q7 - CTS 4.6.1.e.3(b) requires the diesel generator to operate "loaded with emergency loads" for ≥ 5 minutes. ITS SR 3.8.1.9 requires the diesel generator to operate for ≥ 5 minutes, with no specification on loading levels. Discuss the omission of specifying the loading of the diesel generator in this ITS Surveillance. Document the mechanism that controls the diesel generator loading for this test.

Response: The first part of CTS 4.6.1.e.3(b) states that the DG must autostart and energize "the automatically connected emergency loads." This is followed later by a requirement to operate the DG "loaded with emergency loads" for ≥ 5 minutes. The ITS SR 3.8.1.9 requires that DG autostart and energize "automatically connected emergency loads and operate for ≥ 5 minutes." The only difference between the two parts of CTS 4.6.1.e.3(b) (and the CTS and ITS) is that the first part requires "automatically connected emergency loads" while the

second part only specifies "emergency loads." RG&E does not believe there to be a difference between these statements; however, even if it were assumed to be a difference, then the use of "automatically connected emergency loads" is conservative. UFSAR Table 8.3-2 lists all emergency loads supplied by the DG. This table is organized into three categories: (1) loads during the injection phase, (2) loads during the high-head recirculation phase, and (3) loads during the low-head recirculation phase. The loads specified during the injection phase are automatically loaded while the loads during the two recirculation phases may be either automatically or manually loaded. As can be seen from this table, there is never an instance in which all possible emergency loads are being supplied by the DGs. However, the loads during the recirculation phases are significantly less than those during the injection phase where the equipment is automatically loaded. Consequently, RG&E considers the wording of SR 3.8.1.9 to be appropriate. [This response was later revised based on meetings the week of 11/13/95. See comment #212.]

With respect to controlling DG loading, procedure RSSP 2.2, Diesel Generator Load and Safeguard Sequence Test, generates a loss of offsite power and SI signal for each DG and verifies that the DG starts within 10 seconds and that all automatic loads are sequenced within acceptable time limits. This is supplemented by SR 3.8.1.3 which requires a monthly diesel generator load test at maximum expected loads.

3.808 - CTS 4.6.1.e.4, "This test may also serve to concurrently met the requirements of 4.6.1.a and b" (cold shutdown and refueling, and except for cold shutdown and refueling, respectively) is relocated to Note 1 of ITS SR 3.8.1.2, "Performance of SR 3.8.1.9 satisfies this SR." Discuss how this applies during cold shutdown and refueling, including ITS SR 3.8.2.1, "for AC sources required to be OPERABLE." Show the equivalence during cold shutdown and refueling. Discuss and justify any differences.

Response: ITS SR 3.8.2.1 states the following:

For AC sources required to be OPERABLE, the following SRs are applicable:

SR 3.8.1.1 SR 3.8.1.4
SR 3.8.1.2 SR 3.8.1.5

For a DG which is required to be OPERABLE in MODES 5 and 6 (i.e., CTS cold shutdown and refueling), ITS SR 3.8.1.2 must have been performed within the last 31 days. However, ITS SR 3.8.1.2 has a Note which states that "Performance of SR 3.8.1.9 satisfies this SR." Consequently, if ITS SR 3.8.1.9 has been performed within the last 31 days, then ITS SR 3.8.1.2 is considered met and ITS SR 3.8.2.1 is also considered met with respect to this surveillance. RG&E does not believe this is any different than CTS 4.6.1.e.4 which essentially states that the refueling outage based DG test meets the requirements of the monthly DG test during cold shutdown and refueling (CTS 4.6.1.a) and the monthly DG test above cold shutdown and refueling (CTS 4.6.1.b).

3.8Q9 - The CTS 4.6.2.c requirement to trend battery test data is deleted. Justification D.33.x states the trending is "performed to meet the frequency requirements of SR 3.8.6.2 and SR 3.8.4.3." Explain how the trending applies to these surveillances. Describe the trending program and its associated controls.

Response: This response is organized into several parts. First, SR 3.8.6.2 requires verification that battery cell parameters are met every 92 days. These battery cell parameters provide actual acceptance criteria for battery OPERABILITY which if not met, have specific Required Actions to be performed. However, the battery cell parameter limits also provide margin to the absolute OPERABILITY limits per IEEE Std 450-1987 which states that the limits and corrective actions are meant to provide "optimum life of the battery." For example, a battery's electrolyte level is not a critical issue unless the plates are in danger of being exposed. Requiring the electrolyte level to be greater than the minimum water level indication mark on the battery cell provides margin to exposing the plates. Therefore, it can be inferred that these battery parameters perform the same function as trending in that it ensures that batteries remain at their optimum performance. Meanwhile, SR 3.8.4.3 requires verification of battery capacity every 60 months or at an increased frequency due to degradation. In order to measure degradation, trending must be performed. Consequently, SR 3.8.6.2 and SR 3.8.4.3 provide equivalent control to CTS 4.6.2.c.

Second, the trending program consists of the following. After the monthly and quarterly battery checks as required by CTS 4.6.2.a and 4.6.2.b, the Electrical PM Analyst is required to review the data by his signature in procedure PT-11, 60 Cell Battery Banks "A" & "B" and Spare Cells (attached). The Electrical PM Analyst also adds the test results to a data base at which point a data trend can be made. Procedure PT-11 which requires the Electrical PM Analyst's signature, must have an evaluation performed per 10 CFR 50.59 for any change.

3.8Q10 - CTS 4.6.2.d requires a battery load (performance) test every 12 months with a possible extension of 3 additional months. ITS SR 3.8.4.2 requires a battery service test every 24 months. Explain the 'load' test and the 'service' test. Describe and justify any differences. Justify the extended interval between tests.

Response: This response is organized into several parts. First, the only difference between a "load" test and "service" test is the name. Otherwise, the testing requirement remains the same. As a matter-of-fact, the procedures which implement CTS 4.6.2.d are titled "Station Battery 1A Service Test" (Procedure PT-10.3) and "Station Battery 1B Service Test" (Procedure PT-10.2). The purpose of these procedures is to demonstrate that a battery will carry the expected emergency load profile for 4 hours without the battery terminal voltage falling below a specified value.

Second, the justification for the increase in surveillance interval from 12 months to 24 months is as follows. IEEE Standard 450-1987,

Section 5.3 states that the battery service test is required for nuclear applications; however, no testing interval is specified. The EPRI guidance for batteries (Nuclear Maintenance Applications Center, Stationary Battery Maintenance Guide, TR-100248, dated December 1992) specifies a service test frequency of "annually or each refueling outage." A review of plant records since the current batteries were installed in 1986 and 1990 shows that neither battery has failed this service test. Consequently, RG&E believes this surveillance interval to be acceptable, especially with the monthly and quarterly verifications required by ITS SR 3.8.6.1 and SR 3.8.6.2.

3.8Q11 - CTS 4.6.3.a.1 confirms nominal voltage on the high side of transformers 12A and 12B whereas ITS SR 3.8.1.1 confirms "indicated power availability for the offsite circuit to each of the 480 V safeguards buses." Explain how these surveillance requirements accomplish the same objective. Present justification for any differences.

Response: The nominal voltage on the high side of transformers 12A and 12B is not an assumption of any accident analysis. In addition, the power supplied from these transformers is 4160 VAC which must be transformed down to 480 V before it reaches the four safeguards buses. If sufficient voltage is not available on transformers 12A and 12B, then the 480 V safeguards buses would automatically trip and require the associated DG to start and supply the necessary loads. Verifying that indicated power is available to each 480 V safeguards bus per SR 3.8.1.1 ensures that an offsite power source is available and capable of supplying accident loads. In addition, ITS SR 3.8.9.1 requires verification of acceptable voltage on the 480 V safeguards buses. Verifying breaker alignments and 480 V bus voltage ensures that the accident assumptions are met. Requiring verification at the high side of transformers 12A and 12B does not ensure that power is available to the 480 V safeguards buses since breakers, transformers, and buses between transformers 12A and 12B and the 480 V buses could fail.

3.8Q12 - CTS 4.6.3.a.2 verifies 4160-Vac circuit breaker position. Either 12AX or 12BX AND 12AY or 12BY must be open. The referenced ITS SR 3.8.1.1 states "verify correct breaker alignment." Describe and justify this relocation of requirements (the details of the breaker alignment). What are the controls on the relocated requirements?

Response: This response is organized into several parts. First, the CTS 4.6.3.a.2 verification ensures that both offsite sources are not supplying the same 4160 V bus. If this were to happen, a fault could fail both offsite power sources. However, this scenario is bounded by operation in the 100/0 mode (i.e., one offsite power source supplying both buses) in which the plant can backfeed through the main transformer as a secondary power supply. Second, the configuration of these breakers is controlled by procedure 0-6.13, Daily Surveillance Log, Attachment I, page 3 of 5 (attached) which requires verification of breaker positions once a day. This procedure requires an evaluation per 10 CFR 50.59 for any changes.

3.8Q13 - CTS 4.6.3.a.3 verifies that tie breakers 52/BT16-14 and 52/BT17-18 are open when the RCS temperature > 200°F. The referenced ITS SR 3.8.9.1 confirms "correct breaker alignment." Describe and justify this relocation of requirements (the details of the breaker alignment and RCS temperature limit). What are the controls on the relocated requirements?

Response: The tie breaker position is specified in the LCO bases section for LCO 3.8.9 which requires these two breakers, and 3 other AC breakers, to be opened as part of the OPERABILITY requirements for the associated electrical distribution system (see page B 3.8-81 of Attachment C). ITS SR 3.8.9.1 then verifies these correct breaker alignments every 7 days. As such, CTS 4.6.3.a.3 is not changed in the ITS, only the specific listing of the tie breakers is relocated to the Bases under the Bases Control Program. Relocation of this level of detail to the bases is consistent with NUREG-1431 which prevents the need for TS changes when equipment identification numbers change. The addition of 3 new AC breakers ensures that independence is maintained between the two electrical distribution trains.

3.8Q14 - The completion time for LCO 3.8.3, Action A.1 is 48 hours in both the ITS and NUREG-1431. The basis is a 40-hour supply of diesel fuel and a 7-day supply of diesel fuel, respectively. It is not apparent that the time is derived from the current Technical Specifications. Justify the 48-hour completion time to restore a 40-hour supply of diesel fuel. Note: The CTS Basis for 3.7.1 and 3.7.2, page 3.7-5, states "deliveries within 8 hours." Thus, it appears the completion time for Action A.1, to restore the fuel oil level to within limits, should be about 8 hours.

Response: The NUREG-1431 Completion Time of 48 hours for LCO 3.8.3, Required Action A.1 was based on a compromise between the industry and NRC during the development of the NUREG. As noted by the reviewer, NUREG-1431 includes a requirement for a 7 day supply of diesel fuel for each DG while CTS 4.6.1.b.3 (and ITS SR 3.8.3.1) only requires a 40 hour supply. The CTS bases also state that "commercial oil supplies and trucking facilities exist to assure deliveries within 8 hours." While 8 hours may be a more prudent Completion Time, RG&E proposes a Completion Time of 12 hours to ensure that fuel oil can be delivered. A ratio of 12 hours to restore a 40 hour supply is essentially equivalent to the NUREG-1431 ratio of 48 hours to restore a 7 day supply. Comment #30 has been opened to address this.

3.8Q15 - The Background Basis for ITS 3.8.4 states that DC distribution panels A are the normal dc supply for Train A (Buses 14 and 18 and diesel generator A) and the emergency dc supply for Train B (Buses 16 and 17 and diesel generator B) and that DC distribution panels B are the normal dc supply for Train B (Buses 16 and 17 and diesel generator B) and the emergency dc supply for Train A (Buses 14 and 18 and diesel generator A). Discuss how divisional independence is maintained and the surveillance that assures it.

Response: This response is organized into several parts. First, the issue of divisional independence was evaluated by the NRC during the Systematic Evaluation Program (SEP) with a safety evaluation issued on February 21, 1981 (attached). This evaluation acknowledged that the DC distribution system did not meet the current criteria for independence of onsite power systems. However, in the integrated safety assessment of all issues which did not meet current criteria, the NRC only required administrative control of breaker and fuse status as a final resolution (see NUREG-0821, section 4.2.4). Second, the LCO bases for LCO 3.8.9 require the tie breakers to be opened when in MODES 1, 2, 3, and 4 (see page B 3.8-81 of Attachment C). ITS SR 3.8.9.1 then verifies these correct breaker alignments every 7 days.

3.8Q16 - ITS SR 3.8.4.3, battery capacity (performance discharge) test, has a frequency of 60 months (12 months or 24 months for certain conditions). There is also a note prohibiting this surveillance in Modes 1, 2, 3, and 4. It is noted that a 24-month refueling interval is proposed. When a 12 month frequency is required (degraded battery or the battery has reached 85% of expected life and < 100% rated capacity) and the plant is in Mode 1, describe the actions that result in the timely completion of this surveillance. Justify any deviations from the 12-month requirement. In a similar manner, discuss the 60-month interval imposed on the 24-month refueling cycle, that is, the outages to perform the test occur at 24 months, 48 months, and 72 months, and do not correspond to the 60-month interval specified. What is the RG&E method, means, and routine to complete this surveillance prior to 60 months?

Response: This response is organized into several parts. First, if a 12 month surveillance is required which cannot be performed in MODES 1, 2, 3, and 4, the plant must either: (1) shutdown to perform the test, (2) request enforcement discretion, or (3) obtain a technical specification change. The plant cannot continue to operate since if ITS SR 3.8.4.3 is not valid, then SR 3.0.1 requires declaring the affect battery inoperable. Therefore, there cannot be any deviations from the 12 month requirement. However, it should be noted that Ginna Station installed new batteries in 1986 and 1990 such that this degradation imposed surveillance frequency is not expected to be reached.

Second, the 60 month surveillance interval of ITS SR 3.8.4.3 is based on 12 month cycles (i.e., one test every 5 refueling outages). Implementation of 24 month cycles would require this surveillance once every other refueling outage (i.e., at 48 months). Implementation of 18 month cycles would require this surveillance once every third refueling outage (i.e., at 54 months). However, ITS SR 3.8.4.3 only requires that the surveillance be performed at intervals not to exceed 60 months. It does not require the surveillance to be performed at exactly 60 month intervals. In addition, the 60 month Frequency is consistent with NUREG-1431. Therefore, RG&E does not consider this to be an issue.

3.8Q17 - Describe why the T. S. C. vital battery (shown in Figure B 3.8.4-1)

does not need inclusion in LCO 3.8.4, Conditions, Actions, or Surveillance. Otherwise, why is it included in the Figure?

Response: The Technical Support Center (TSC) vital battery is included on the figure "for information only" to provide a reference as to how this battery source can be utilized if required (e.g., a beyond design basis station blackout event). Plant operators requested the TSC battery to be included on the figure for completeness since it is included in their training material. The TSC battery is not credited in any accident analysis, and as such, does not require any Conditions, Actions, or Surveillances.

3.8Q18 - Figure B 3.8.4-1 shows 5 circuit breakers that are "normally open when $T_{avg} > 200^{\circ}F$." Where is the associated surveillance that verifies the breaker alignment? If not located in LCO 3.8.4, DC Sources, why not? If controlled by LCO 3.8.9 and 3.8.10, describe why those controls are suitable in maintaining the independence of the sources.

Response: This response is organized into several parts. First, the tie breakers for the AC and DC power systems which must be opened when in MODES 1, 2, 3, and 4 are specified in the LCO bases section for LCO 3.8.9 (see page B 3.8-81 of Attachment C). ITS SR 3.8.9.1 verifies these correct breaker alignments every 7 days. Second, of the five tie breakers shown in Figure B 3.8.4-1 which must be opened, only the two tie breakers between 480 VAC Buses 14 and 16 are identified in the bases for LCO 3.8.9. The three tie breakers related to the TSC battery are not listed since the TSC battery cannot be credited as a battery source when in MODES 1, 2, 3, and 4. Third, the acceptability of the controls for the AC and DC distribution system is provided in a NRC safety evaluation dated February 21, 1981 (attached). The AC distribution system was found to meet current requirements while the DC distribution system is discussed in the response to 3.8Q15.

3.8Q19 - Describe why the Specific Gravity limits of ITS Figure B 3.8.6-1 are different for Battery A and Battery B. Describe the validity of the limits presented in ITS Figure B 3.8.6-1.

Response: This response is organized into several parts. First, the specific gravity limits for the batteries are based on the initial specific gravity limits following installation. This is in accordance with IEEE Std 450-1987 which requires an equalizing charge if the average specific gravity of all cells drops more than 10 points from the "average installation value" (see Section 4.4.2). The subject batteries were installed in 1986 and 1990 with slightly different originally measured specific gravity values. Consequently, the specific gravity limits are different for the two batteries. It is for this reason that RG&E has proposed to relocate NUREG-1431 Table 3.8.6-1 from the LCO since it could potentially require a technical specification change to replace a battery with the exact same design. However, the RG&E proposed fix does not completely solve this issue since ITS LCO 3.8.6 directly references bases Table B 3.8.6-1. Comment #4 has been opened to address this.

Second, the basis for the remaining values of ITS Figure B 3.8.6-1 is as follows:

- a. *Electrolyte Level* - The proposed wording is consistent with NUREG-1431 Table 3.8.6-1. IEEE Std 450-1987 only specifies adding water if the low level line is reached. Therefore, this wording is considered acceptable.
- b. *Float Voltage* - The proposed value (2.13 V) is consistent with NUREG-1431 Table 3.8.6-1. IEEE Std 450-1987 requires an equalizing charge if any cell is below 2.13 V (see Section 4.4.3). Therefore, this value is considered acceptable.

3.8Q20 - ITS B 3.8.7 states that the loss of Instrument Bus D is addressed in LCO 3.3.2 and LCO 3.3.3. The basis for LCO 3.3.3 tells of the affected instrumentation and the need to declare them inoperable. However, neither LCO 3.3.2 nor LCO 3.3.3 direct actions on the loss of Instrument Bus D. Neither does LCO 3.8.7. Describe why this omission is acceptable.

Clarify whether the ITS is intended to cover three or four 120-Vac Instrument Buses. B 3.8.9, Background, states the "AC Instrument Bus electrical power distribution subsystem consists of four 120 VAC instrument buses." Thus, LCO 3.8.9, Condition B, includes Instrument Bus D. However, it is not clear whether LCO 3.8.7, Condition D, includes MCC B, the power source for Instrument Bus D. The ITS appears inconsistent on the application and use of 120-Vac Instrument Bus D. The scope of the Instrument Bus requirements should be consistent, that is, either include 120-Vac Instrument Bus D throughout or don't include it. This concern may require a telephone conference to clarify and resolve.

Response: This response is organized into several parts. First, Instrument Bus D as shown in ITS Figure B 3.3.2-1, is supplied by a non-diesel generator backed bus (MCC B). Therefore, upon loss of offsite power, this Instrument Bus is unavailable and is not included in LCOs 3.8.7, 3.8.9 or 3.8.10. The bases for these three LCOs state that the need for Instrument Bus D is instead addressed in LCOs 3.3.2 and 3.3.3. In addition, Table 3.8.9.1 does not list Instrument Bus D as being included in the AC and DC Electrical Power Distribution Systems. RG&E believes this provides sufficient information of which LCOs to consider if Instrument Bus D is unavailable.

Second, there is one ESFAS function (LCO 3.3.2) and two PAMS functions (LCO 3.3.3) which are partially supplied power via Instrument Bus D. Consequently, the availability of Instrument Bus D directly affects these function's OPERABILITY. If this instrument bus is inoperable, then the affected instrumentation must be declared inoperable. This is discussed on ITS bases pages B 3.3-109 (first paragraph), B 3.3-109 (last sentence) and B 3.3-120 (first sentence) with respect to PAMS. Similar wording on bases page B 3.3-120 should also be provided with respect to the AFW initiation on low SG level (i.e., the same SG level transmitters affected by the loss of Instrument Bus D in PAMS are also affected in ESFAS).

Comment # 31 has been opened to address the ESFAS omission.

Third, the CTS have no OPERABILITY requirements for Instrument Bus D which RG&E believes is acceptable since it is supplied by a non-safety related power source. Ginna Station operators believe that the reference to LCOs 3.3.2 and 3.3.3 in LCOs 3.8.7, 3.8.9, and 3.8.10 provides the necessary information to ensure appropriate actions are taken if Instrument Bus D is unavailable. Providing required actions when only three ITS functions are affected is unnecessary. Also, surveillances of Instrument Bus D are not required since the affected instrumentation already have appropriate surveillances. In addition, if Instrument Bus D were lost, a reactor trip could potentially occur due to affected components (e.g., pressurizer control).

3.8Q21 - The ITS Basis for SR 3.8.9.1 and SR 3.8.10.1 list the requirements for the AC Instrument Bus power distribution subsystem as "between 113 VAC and 123 VAC." The "required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC," and is more restrictive than the limits on the instrument bus voltage. Describe the use of the twinco panels and justify why the technical specifications should limit the voltage maintained there to a value more restrictive than the voltage limits on the power source to the panels.

Response: *The instrument buses supply power to the twinco panels which in turn supply the safety-related instrument loops. Both the twinco panels and instrument buses were purchased and installed with an allowed tolerance of $\pm 2\%$ voltage. Analyses have been performed which demonstrate that loads directly supplied by the instrument buses can withstand tolerances greater than $\pm 2\%$. However, due to instrument sensitivity concerns related to the loads supplied by the twinco panels, these panels are limited to $\pm 2\%$ of 118 VAC.*

3.8Q18A The tie breakers for the ac and dc power systems that must be open in Modes 1, 2, 3, and 4 are specified in the Bases for LCO 3.8.9. Improved Technical Specification Surveillance Requirement 3.8.9.1 verifies these correct breaker alignments every 7 days. Of the five tie breakers shown in Figure B 3.8.4-1 that must be open, only the two tie breakers between 480-Vac Buses 14 and 16 are identified in the bases for LCO 3.8.9. The three tie breakers related to the Technical Support Center vital battery are not listed since the Technical Support Center vital battery cannot be credited as a battery source when in MODES 1, 2, 3, and 4. Detail why these tie breakers should not be verified open by technical specification in support of the independence required by Regulatory Guide 1.75?

Response: *Verification of the TSC vital battery tie breakers has been added to the bases of LCO 3.8.9.*

3.8Q21A The ITS Basis for SR 3.8.9.1 and SR 3.8.10.1 list the requirements for the AC Instrument Bus power distribution subsystem as "between 113 VAC and 123 VAC." The "required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC," and is more restrictive than the limits on the

instrument bus voltage. The instrument buses supply power to the Twinco panels which in turn supply the safety-related instrument loops. Both the Twinco panels and instrument buses were purchased and installed with an allowed tolerance of $\pm 2\%$ voltage. Due to instrument sensitivity concerns related to the loads supplied by the Twinco panels, these panels are limited to $\pm 2\%$ of 118-Vac. By use of the $\pm 2\%$ limit on the Twinco panels, that same limit is effectively on the ac instrument buses also. Shouldn't the Bases be changed to the Twinco $\pm 2\%$ limit (or an offset $\pm 2\%$ to account for voltage drop in the distribution system)?

Response: To be discussed at the meeting.

3.8QA2 Supply the number of 'pilot' cells for the Category A measurements of SR 3.8.6.1. Supply the number of 'representative' cells for SR 3.8.6.3. What is the difference between 'pilot' cells and the 'representative' cells are determined, assigned and whether they remain as such throughout the life of the battery.

Response: To be discussed at the meeting.

3.8QA1 Attachment A, Section C, Conversion of NUREG-1431-1431 to ITS, Item 94.vi, on ITS 3.8.1 notes that NUREG-1431 SR 3.8.1.8 (ITS SR 3.8.1.6), SR 3.8.1.10 (ITS 3.8.1.7), and SR 3.8.1.13 (ITS SR 3.8.1.8) were revised to restrict performance of these SRs in Modes 3 and 4. The draft ITS SR 3.8.1.6 does not have that restriction on performing these SRs which would reduce the number of available ac sources and could cause perturbations to the electrical distribution system and challenge safety systems. Correct ITS SR 3.8.1.6 to include this restriction.

Response: To be discussed at the meeting.

Section 3.9 Current TS

15. viii [CTS 15.viii-R1]
TS 3.5.5 and Table 3.5-5 - The requirements for radioactive effluent monitoring instrumentation which ensures that the limits of TS 3.9.1.1 and 3.9.2.1 are not exceeded were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable or, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively.

3.9Q1 Explain how the ITS conform to the guidance of generic letter 89-01 to incorporate programmatic controls for radioactive effluents and radiological and environmental monitoring consistent with the requirements of 10 CFR Part 20, 40 CFR Part 190, and 10 CFR Part 50 Appendix I.

Response: The application of the NRC Final Policy Statement Technical Specification screening criteria allows for the relocation of the Radiological Effluent Technical Specifications from the CTS. These changes are consistent with the guidance provided in GL 89-01. The relocation of requirements and subsequent incorporation of associated programmatic controls in the ITS conforms to the regulatory requirements of 10 CFR Part 20, 40 CFR Part 190, and 10 CFR Part 50, Appendix I. The ITS Chapter 5.0, "Administrative Controls" provides the programmatic controls necessary to ensure that the programs are established, implemented, and maintained to provide conformance with the regulatory requirements.

3.9Q2 *Submit any additional information required to meet the guidance of GL 89-01. Provide a markup of the CTS specifying which details of the CTS are to be relocated to the ODCM, effluent controls program, ITS specification 5.5.1 or ITS specification 5.5.4.*

Response: The GL 89-01 requests that three items be submitted in the license amendment request:

First, GL 89-01 required that the "model" TS be included. These TS were based on the Standard TS which formed the basis for the Improved Standard TS. Any technical change from the ITS is provided with a specific justification in the May submittal; therefore, no further information is required.

Second, GL 89-01 required that if changes (other than editorial changes) were made to the procedural details relocated from the CTS to the ODCM or Effluent Controls Program, then these changes were to be identified in the license amendment request. There are no changes to the relocated procedural details; therefore, no further information is required.

Third, GL 89-01 required confirmation in the license amendment request that the ODCM was revised and reflected the relocated requirements. This was to ensure that the relocated requirements could be implemented immediately upon issuance of the license amendment request. The GL 89-01 required that a complete and legible copy of the revised ODCM be forwarded with the license amendment request. The GL stated that the ODCM was being submitted for reference purposes only and that the NRC Staff would not concur in or approve the revised ODCM document. RG&E believes that submittal of the ODCM with the ITS conversion package is not necessary since relocation of the Radiological Effluent Technical Specifications from the CTS were based on utilizing the NRC Final Policy Statement Technical Specification screening criteria. Therefore the relocation of these requirements, as well as all other relocated requirements beyond the scope of GL 89-01, only required denoting the appropriate licensing document location (such as the TRM, ODCM, UFSAR, or ITS Bases). Moreover, the implementation of the ITS requires that all relocated requirements be effectively incorporated in their associated plant program or operating procedures upon issuance of the ITS.

The CTS were marked addressing Radiological Effluent Technical

Specifications as part of changes addressed in Chapter 5.0. Therefore, no further information is required.

3.9Q3 What is the method of control for approving changes to the effluent controls program after implementing ITS?

Response: The method of control for approving changes to the effluent controls program after implementing ITS is described in Specification 5.5.1 of the ITS.

3.9Q4 Clarify the meaning of the margin notation "shutdown purge and minipurge in Mode 6, others are addressed w/ Chapter 5.0" of page 3.5-2 as it related to discussion 15.viii]

Response: The right-hand margin notation was intended to state that only the shutdown purge and mini-purge instrumentation was to be addressed with respect to ITS Chapter 3.9 while the remaining instrumentation is addressed in Chapter 5.0. However, the left-hand margin should not reference change 15.viii. Instead, it should reference a new change 15.ix which states:

The requirements for the shutdown purge and mini-purge instrumentation were not added to the new specifications since these functions are not credited in the accident analyses. In MODES 1, 2, 3, and 4, only containment isolation is credited with respect to isolating containment. In MODE 6, containment isolation is not credited at all. Attachment A, Section D, items 15.ii.p and 18.i provide additional information. Therefore, these requirements were relocated to the TRM. This is a Ginna TS Category (iii) change.

Comment #109 has been opened to correct the Attachment B markup and to add the above change to Attachment A. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #221.]

16.x

[CTS 16.x-A1/L1]

TS 3.6.1.b and TS 3.6.1.c - The requirement describing the specific applicability for containment integrity was not added. No screening criteria apply for this requirement since containment integrity is not assumed in the refueling safety analysis. The fuel handling accident inside containment analysis (UFSAR 15.7.3.3) takes no credit for isolation of the containment, containment integrity, nor effluent filtration prior to release. The requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. Boron concentration changes in MODE 6 and Required Actions to suspend positive reactivity additions is provided in new LCO 3.9.1. This is a Ginna TS Category (iii) change.

Assumptions of the evaluation of the fuel handling accident inside containment (Attachment A, Reference 49 p.3, para. 2) state that TS require that personnel and equipment doors are closed. Further, the NUREG-1431 refueling operations containment penetrations LCO 3.9.4 satisfies criterion 3 of 10 CFR 50.36.

3.9Q5 Current TS 3.6.1.b requires containment integrity with the vessel head removed unless boron concentration is greater than 2000 ppm. Discuss how TS 3.6.1.b requirements are provided in refueling ITS LCO 3.9.1. What part of TS 3.6.1.b is proposed to be relocated?

Response: ITS LCO 3.9.1 requires that the boron concentration of the RCS shall be maintained within the limit specified in the COLR (2000 ppm) during MODE 6 which is defined in ITS Table 1.1-1 as any time "one or more reactor vessel head closure bolts [are] less than fully tensioned." Therefore, the MODE of Applicability for ITS LCO 3.9.1 bounds "when the reactor head is removed" as specified in CTS 3.6.1.b. In addition, the Required Actions in the event that boron concentration limits are not met in the ITS are to suspend CORE ALTERATIONS and positive reactivity additions consistent with CTS 3.6.1.c. These actions effectively prevent the two accidents of concern in MODE 6 (i.e., a fuel handling accident and boron dilution event). The containment isolation issue is addressed in the response to 3.9Q7 below. Therefore, the CTS 3.6.1.b requirement for boron concentration limits being > 200 ppm when the reactor head is removed has been relocated to ITS LCO 3.9.1.

3.9Q6 Current TS 3.6.1.c requires containment integrity and prohibits reactivity changes unless boron concentration limits are met. Discuss how TS 3.6.1.c requirements are provided in ITS LCO 3.9.1. What part of TS 3.6.1.c is proposed to be relocated? Discuss how containment integrity requirements of TS 3.6.1.c can be added to the LCO 3.9.4 or to a section 3.6 TS applicability since TS 3.6.1.c establishes limits related containment closure prior to boration to 2000 ppm.

Response: See response to 3.9Q5. Essentially, the Required Actions for ITS LCO 3.9.1 require suspension of CORE ALTERATIONS and positive reactivity additions whenever the boron concentration limit is not met in MODE 6 regardless of the containment status. Therefore, CTS 3.6.1.c is being relocated in its entirety to ITS LCO 3.9.1.

18. Technical Specification 3.8

- i. [18.i-XX]
TS 3.8.1.a and 3.8.3 - The requirements to close containment penetrations during fuel handling in the containment were not added. No screening criteria apply for these requirements since these conditions are not assumed in the refueling safety analysis. The fuel handling accident inside containment analysis (UFSAR 15.7.3.3) takes no credit for isolation of the containment nor effluent filtration prior to release from the containment building. Therefore, closure of containment penetrations during fuel handling inside containment is not required. The closure of the containment penetrations were established to provide additional margin for the fuel handling analysis and to provide protection against the potential consequences of seismic events during refueling. The dose consequences, however, of the fuel handling accident inside containment analysis is estimated at approximately 30% of 10 CFR 100 limits. This was found to be "well within" limits as documented in the NRC Safety Evaluation Report (SER) dated October 7, 1981 (Ref.

49). The requirements specified for these conditions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

3.9Q7 In the evaluation of the consequences fuel damaging accidents inside containment (SEP topic XV-20) dated October 7, 1981 the staff assumed that the plant TS require that personnel and equipment doors be closed and radiation levels be continuously monitored (page 3, paragraph 2). The assumptions of this evaluation continue to be valid. Provide a markup of NUREG LCOs 3.9.4, 3.9.5 and 3.9.6 and associated Bases pages with appropriate justifications showing incorporation of existing TS 3.8.1.a and 3.8.3.

Response: The October 7, 1981 SER calculated a dose of 96 rem at the EAB assuming that the personnel and equipment doors were closed and that radiation levels were continuously monitored. This dose was stated to be "well within the guideline value of 10 CFR Part 100." RG&E subsequently requested that the NRC calculation details be provided (see attached letter dated November 4, 1981). The NRC replied that these details were unnecessary since the "methods employed are contained in the Standard Review Plans" and that all input and assumptions were included in the safety evaluations (see attached letter dated March 3, 1992). As such, RG&E performed its own analysis of the fuel handling accident as contained in UFSAR Section 15.7.3.3 (attached). This UFSAR analysis assumes that no containment isolation is provided, including the personnel and equipment doors, and calculates a dose of 103 rem at the EAB. Based on conversations with Westinghouse, these assumptions are consistent with older, smaller plant designs such as Ginna. This UFSAR analysis provides the bases used in the development of the Ginna ITS. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #221.]

ii. [18.ii-R1]
TS 3.8.1.b - The refueling or MODE 6 requirement for the containment radiation monitors which provide monitoring for personnel safety was not added. No screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the containment radiation monitors are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

3.9Q8 Provide justification that the containment radiation monitors are not required to prevent the possibility of an abnormal situation or event giving rise to an immediate threat of the public health and safety.

Response: The CTS 3.8.1.b requires the continuous monitoring of radiation levels. The monitoring requirement is not associated with any required automatic isolation function and is provided for personnel



safety only. The accident analyses also do not credit manual operator action through use of these monitors since there is no design basis event that credits this function. Therefore, there is no abnormal situation or event giving rise to an immediate threat of the public health and safety.

iii. [18.iii-A1/M1]

TS 3.8.1.c - The requirement describing the specific applicability of the SRMs was revised. The phrase "whenever geometry is being changed" is covered by the new TS definition of MODE 6. [A] The requirement that one SRM be OPERABLE when core geometry "is not being changed" is covered by the Required Action [3.9.3 RA A.1 and A.2] for one inoperable SRM. [A] This would restrict CORE ALTERATION and positive reactivity additions when core geometry is not being changed. Required Actions 3.9.3 Conditions B and C] were also provided when two SRMs become inoperable or when the audible indication is lost. [M] These new actions require verification of boron concentration every 12 hours and ensures the stabilized condition of the reactor core. These are a conservative revisions and Ginna TS Category (v.a) and (iv.a) changes, respectively.

3.9Q9

Provide an explanation justifying the conclusion in 18.iii that the proposed ITS are more restrictive "technical" and more restrictive "additions" to the requirements in TS 3.8.1.c. Specifically identify these Ginna TS Categories for each proposed change. Justify each conclusion.

Response:

The CTS 3.8.1.c requires one operable SRM during MODE 6 when core geometry is not being changed and two operable SRMs during refueling when the core geometry is being changed. The ITS LCO 3.9.2 requires two SRMs operable at all times during MODE 6. The ITS LCO represents the lowest functional capability or performance levels of equipment required for safe operation of the facility. Therefore, the ITS has been developed requiring two SRMs operable. With only one SRM operable, the Required Actions A.1 and A.2 are no more limiting than what is specified by the CTS LCO requirement (i.e., no fuel movement is allowed). The change is considered more restrictive only because the ITS places the plant in a specific Condition whereas the CTS LCO would continue to be met. As such, this is a "v.a" change.

The CTS also requires the suspension of operations which may increase core reactivity. The CTS has been revised to add the requirements denoted by LCO 3.9.2 Required Actions B1, B4, and C.3 when two SRMs become inoperable or when the audible indication is lost. These new actions require the immediate initiation of action to restore one SRM to operable status and verification of boron concentration every 12 hours. These Required Actions ensure the stabilized condition of the reactor core. Because these Required Actions are new requirements, this change is considered more restrictive with respect to CTS (i.e., a "iv.a" category change).

iv. [18.iv-A1/M1/L1]

TS 3.8.1.e - The requirement describing the specific applicability

and frequency of the boron concentration sampling was revised. The phrase "immediately before reactor vessel head removal and while loading and unloading fuel from the reactor" is covered by the new TS definition of MODE 6. [A1] This would additionally require boron concentration sampling throughout MODE 6. [M1] The sampling frequency, however, was also revised [from requiring boron sampling twice each shift] to require sampling every 72 hours. [L1] These revisions consider the large volume of the refueling canal, RCS, and refueling cavity and are adequate to identify slow changes in boron concentration. Rapid changes in boron concentration, described in UFSAR 15.4.4.2, are detected by the SRM instrumentation required by new TS 3.9.2. This is a conservative revision and a Ginna TS Category (iv.a) change.

3.9Q10 Provide an explanation justifying the conclusion in 18.iv that the proposed ITS are more restrictive than the requirements in TS 3.8.1.e. Justify that the proposed changes are enhancements to the existing TS. Another proposed change addresses the applicability and frequency of boron sampling. State why there is not a significant safety question in the operation of the plant by changing the frequency of existing TS 3.8.1.e to 72 hours from twice each shift.

Response: The CTS 3.8.1.e requires that prior to reactor head removal and while loading and unloading fuel from the reactor, the minimum boron concentration is maintained. The ITS LCO 3.9.1 requires that the minimum boron concentration be maintained during MODE 6. The ITS defines MODE 6 as "when one or more reactor vessel head closure bolts [are] less than fully tensioned." The ITS also requires the SR to be met prior to entering MODE 6 per LCO 3.0.4. Therefore, the CTS requirement to ensure that the minimum boron concentration is maintained "prior to reactor head removal" is equivalent to the ITS requirements. However, the CTS only requires the minimum boron concentration requirement when loading and unloading fuel. The ITS requires continuation of the minimum boron concentration requirement throughout MODE 6. The literal reading of the CTS could be inferred to exclude, as a TS requirement, the requirement to maintain the minimum boron concentration when fuel is not in the process of being loaded or unloaded. Since this requirement could be inferred to be controlled administratively, the ITS requirement is considered more restrictive than that of the CTS.

The CTS 3.8.1.e requires the boron concentration limit be verified twice each shift. The ITS SR 3.9.1.1 requires the boron concentration limit be verified every 72 hours. There is not a significant safety question in the operation of the plant by changing the frequency from twice each shift to 72 hours. The boron concentration limit ensures that the reactor remains subcritical during MODE 6. The boron concentration limit is based in part on the assumptions that: (1) control rods and fuel assemblies are in the most adverse configuration (least negative reactivity) allowed by plant procedures, and (2) core reactivity is at the beginning of each fuel cycle. These conservatisms, along with the fact that the operator has prompt and definite indication in the control room of

a significant boron dilution event, provide assurance that the proposed change in the surveillance frequency does not provide a significant safety question in the operation of the plant. The 72 hour frequency is consistent with NUREG-1431 and is based on industry operating experience which has shown that 72 hours to be adequate.

v. [18.v-R1]

TS 3.8.1.f - The requirement for communication with the control room during CORE ALTERATIONS is not added. No screening criteria apply for this requirement since communications is not part of the primary success path assumed in the mitigation of a DBA or transient. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

3.9Q11 Specify the document to which the TS 3.8.1.f requirements will be relocated and the control mechanism that will be used for making future changes to the requirements.

Response: The requirement is proposed to be relocated to Ginna Station procedure 0-15.1 (attached). Any subsequent changes to this requirement is provided in accordance with the Ginna procedure change process which requires, as minimum, a screening with respect to 10 CFR 50.59.

vi. [18.vi-A1]

TS 3.8.1.d (footnote *) and TS 3.8.1.g (footnote *) - The requirement that either the preferred or the emergency power source may be inoperable for each residual heat removal loop is not added. This detail is encompassed in the definition of operability described in new TS 1.1 and the electric power requirements contained in Chapter 3.8. This is a Ginna TS Category (i) change.

vii. [18.vii-L1]

TS 3.8.1.c - The requirement to provide SRM audible indication in the containment was not added. No screening criteria apply for this requirement since the monitored parameter (audible indication in containment) is not assumed in the refueling safety analysis. The safety analysis assumes audible indication in the control room which is denoted by new LCO 3.9.2. The audible indication is for personnel safety only. Further, the audible indication is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

3.9Q12 Provide a safety justification for why audible indication of neutron flux in the containment is not part of channel operability for the monitors required to be operable by ITS 3.9.2.

Response: The audible count rate function for the containment is provided by

a audible monitor or speaker that receives the same output signal as the audible monitor in the control room. The ITS only requires continuous audible monitoring in the control room. The audible monitoring in the containment is for personnel safety only. No credit is taken for this function in the Ginna safety analysis for the mitigation of any accident. Also, the audible monitor in the containment does not impact the channel OPERABILITY, as defined in the Bases for LCO 3.9.2, for the monitor located in the control room.

3.9Q13 Provide a safety justification for why continuous visual indication of neutron flux in the control room is not included as part of channel operability for the monitors required to be operable by ITS 3.9.2.

Response: The LCO bases for LCO 3.9.2 denotes that channel operability includes visual indication in the control room for each of the channels and an audible count rate function in the control room for one of the two channels. Relocation of this level of detail to the bases is consistent with the rest of the ITS and NUREG-1431.

38. iv. [38.iv-L1]
TS 4.11.2.1 - This was revised to only require verification of RHR pump OPERABILITY once every 12 hours versus 4 hours consistent with SR 3.9.3.1. A Frequency of 12 hours is adequate due to the alarms and indications available to the operators with respect to RHR pump and loop performance. This is a Ginna TS Category (v.b.41) change.

3.9Q14 This LCO verifies loop operability and pump operation. State why there is not a significant safety question in the operation of the plant in changing from 4 hours to 12 hours because of the installed control room alarms and indications.

Response: The CTS 4.11.2.1 requires the verification once per 4 hours that the RHR loop is in operation and circulating water. The ITS SR 3.9.3.1 and SR 3.9.4.1 requires this verification every 12 hours. The change in the frequency from 4 hours to 12 hours is based on the consideration that flow, temperature, pump control, and alarm indications are available to the operator in the control room for monitoring the RHR System. The purpose of the RHR System in MODE 6 is to remove decay heat from the RCS and to provide mixing of the borated coolant to prevent thermal and boron stratification. The SR only requires the verification, through indication in the control room, that the RHR loop is in operation. This SR is redundant to many other indications which would alert the operators should an inadvertent loss of RHR loop occur. The change in frequency is not a significant safety issue since the indication available and the operators response to a loss of RHR loop has not changed. Moreover, LCO 3.9.3 allows the RHR loop to be removed from operation during short durations. These short durations will not result in challenges to the fission product barrier or in coolant stratification since decay heat is removed by natural convection to the large mass of water in the refueling cavity.

v. [38.v-L2, M1]

TS 4.11.2.2 - This was revised to remove the requirement for an Inservice Test of the RHR pumps. An Inservice Test should not be required for an operating pump. [L] The status of a non-operating RHR pump is assured by new SR 3.9.4.2 which requires the verification of the breaker alignment and indicated power available to the pump. [A] The Inservice Testing program test is mainly performed to ensure adequate performance during accident conditions which far exceeds the requirements during normal conditions. This test is not necessary to ensure OPERABILITY during MODE 6 operations. This is a Ginna TS Category (v.b.42) change.

3.9Q15 This LCO verifies pump operability. State why there is not a significant safety question in the operation of the plant in deleting the pump surveillance specified in 10 CFR 50.55a. Provide an administrative change discussion for adding SR 3.9.4.2. Justify that the proposed changes are enhancements to the existing TS.

Response: The CTS 4.11.2.2 requires operability of the RHR pumps by the performance of a pump surveillance specified in 10 CFR 50.55a. The ITS SR 3.9.4.2 requires only verification of correct breaker alignment and indicated power available for the non-operating pump. The specific testing requirement is actually being relocated, and not deleted from the TS, and will be retained in the IST program. The relocation of this requirement is consistent with NUREG-1431 and is administratively controlled within the requirements specified in ITS Chapter 5.0, "Administrative Controls" for the IST program. There is no significant safety question in the operation of the plant, with the relocation of this requirement, since the ITS provides adequate verification that a second RHR pump can be placed in operation to maintain decay heat removal and reactor coolant circulation. Additionally, in MODE 6, the RHR system is not required to mitigate any events or accidents evaluated in the safety analysis. It is only required to provide mixing of the borated coolant to help prevent boron dilution events. A significant amount of time exists before boiling of the coolant could result from a loss of the RHR pumps.

vi. [38.vi-L3]

TS 4.11.3.1 - This was revised to only require a verification of the water level in the reactor cavity within 24 hours of fuel movement versus 2 hours. The new TS usage rules state that a SR is to be continuously performed at its required Frequency. However, the SR is only required to be performed when in the MODE of Applicability. Therefore, a SR with a Frequency of 24 hours must have been performed within 24 hours before entering the MODE of Applicability. A Frequency of 24 hours is acceptable due to the large volume of water available and the procedural controls in place. This is a Ginna TS Category (v.c) change.

3.9Q16 This LCO verifies water level in the reactor cavity. State why there is not a significant safety question in the operation of the plant by changing the frequency of existing TS 4.11.3.1 to 24 hours from 2 hours in proposed SR 3.9.5.1.

Response: The CTS 4.11.3.1 requires verification of water level: (1) within 2 hours prior to the start of the movement of fuel assemblies or control rods in containment, and (2) once per 24 hours thereafter. The ITS SR 3.9.5.1 requires verification of water level: (1) within 24 hours prior to the start of the movement of fuel assemblies in containment (in accordance with SR 3.0.4), and (2) once per 24 hours thereafter. The only difference between the two requirements is the time period of the initial performance of the SR that ensures that the SR is met prior to entry into a specified condition as defined by the Applicability. The ITS SR 3.0.4 ensures that system and component operability requirements and variable limits are met before entry into a MODE or other specified condition (i.e., such as during movement of irradiated fuel assemblies within containment as denoted in the Applicability for LCO 3.9.5) for which these requirements ensure safe operation of the plant. The change has no safety impact on the operation of the plant since in both cases, the water level meets the requirements of the LCO prior to the mode change. This water level is not expected to change between the additional 22 hours which would now be allowed due to administrative controls over valve positions, control room alarms and indications, and operators and other plant personnel typically being inside containment during this time period who could observe a significant change in water level.

104. ITS 3.9.1

3.9Q17 Refer to Bases markup for identification of editorial comments

Response: To be discussed at the meeting.

- i. Incorporation of approved Traveller BWOG-03, C.6.
- ii. Incorporation of approved Traveller WOG-05, C.1. This traveller was modified to provide various wording changes to improve the readability and understanding of the bases. This is an ITS Category (iv) change.
- iii. The LCO was revised consistent with similar LCOs. The details associated with the LCO were relocated to the bases. Including these details in the Bases eliminates ambiguities (e.g., when the plant initially enters MODE 6 and when RCS loops are isolated). This is an ITS Category (iii) change.

3.9Q18 Explain the basis for concluding that the details of the systems in LCO 3.9.1 are not necessary to establish safety basis for systems requiring COLR boron concentration limits. Also, discuss what is meant by ambiguities in the justification. Is there any operational hardship associated with the ambiguity?

Response: The ITS LCO as presented in the NUREG would require the verification of boron concentration limits for the RCS, refueling canal, and the refueling cavity when in MODE 6. This is confusing since the LCO should only apply to the filled portions of these areas that are hydraulically coupled to the reactor core during refueling. For example, UFSAR Figure 9.1-1 (attached) shows a sketch of the reactor



vessel, reactor cavity and refueling canal. The refueling canal can be isolated from the reactor vessel and reactor cavity during MODE 6 by use of a blind flange and gate valve. However, this LCO would still require the refueling canal boron concentration to be the same as that within containment. Instead, the refueling canal boron concentration is addressed by ITS LCO 3.7.12 in this instance. This issue was brought up by Ginna operations during their review of this LCO. This detail is clarified in the Bases but could cause future mis-interpretations to this LCO or require the need for a future license amendment request to this LCO. In order to eliminate this potential confusion, it was proposed to relocate the details in the LCO to the Bases. This is consistent with the details of other TS being relocated to the Bases. Eliminating ambiguity significantly reduces operational hardships due to potential operator errors and reduces training expenses which result from the need for additional clarification. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #195.]

- iv. The bases were revised as follows (these are ITS Category (iv) changes):
- a. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other bases sections.

3.9Q19 Background discussion, page B 3.9-1
Provide replacement discussion applicable to Ginna.

Response: All changes to the NUREG Background Bases are replaced with equivalent text and information. The only exception is with respect to the end of the third paragraph. RG&E agrees to add the previously deleted text beginning with "from the refueling water..." Comment #110 has been opened to address this.

3.9Q20 Actions discussion introduction, page B 3.9-3.
STET the proposed deletion of the first sentence in the paragraph.

Response: This discussion was replaced to more accurately reflect the discussions provided in the "Applicable Safety Analysis" section of this Bases. The change also provides consistency with the rest of the ITS and NUREG bases for Required Actions which have a format as follows:

"If [a specific condition] exists, the plant is outside the accident analysis assumptions and must be restored to OPERABLE status within [completion time]."

The NUREG bases for this section state that CORE ALTERATIONS and positive reactivity additions can only be made if the LCO is met and that if the LCO is not met, these activities must stop. Therefore, the proposed change provides consistency with the rest of the ITS without revising any specific requirements.

- b. The plant-specific description of the boron dilution event during refueling was added.

- c. Ginna Station was designed and built prior to the issuance of the GDC contained in 10 CFR 50, Appendix A. However, the draft GDC issued by the Atomic Industrial Forum (AIF) in 1967 were utilized in the design of Ginna Station. The bases were revised to reflect this difference.

3.9Q21

Bases Insert 3.9.2.

This discussion provides high level design requirement information. Provide background discussion about system capability, i.e., the same level of detail you propose to delete.

Response:

The Insert 3.9.2 appears to be of equivalent level of discussion for the replaced NUREG Background information. NUREG-1431 discusses the CVCS being capable of maintaining the reactor "subcritical" whereas the proposed Bases discusses the CVCS being capable of providing "reactivity control." Additionally, for Ginna, the NUREG-1431 reference to CVCS providing subcritical capability under cold conditions is not appropriate with respect to AIF GDC 27 as referenced in this paragraph. This AIF GDC does not mention cold shutdown requirements at all. However, AIF GDC 29 Criterion states "one of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn." This AIF GDC only relates to "operating conditions" and not "cold conditions" such that this deleted text is not applicable to the Ginna design basis.

- d. The text was revised to clarify that normal cool down of the coolant volume for the purposes of temperature control is not considered as an addition of positive reactivity.
- e. The text was revised to clarify that the sample taken is representative of the RCS, the refueling canal, and the refueling cavity. Only one sample is required since operation of the RHR assumes uniform mixing.

105. ITS 3.9.2

- i. LCO 3.9.2, "Unborated Water Source Isolation Valves" and associated Bases, as presented in NUREG-1431, were not added and subsequent LCOs and Bases have been renumbered in consecutive order. Ginna Station has a plant specific safety analysis (UFSAR Section 15.4.4.2) for an uncontrolled boron dilution event during refueling assuming the worst case scenario with the maximum number of pumps and flow paths available. The conclusion of the analysis establishes that operators have sufficient time (i.e., greater than 30 minutes required by Section 15.4.6 of Reference 3.9.1) to mitigate the effects of a boron dilution event in MODE 6 prior to a loss of SHUTDOWN MARGIN. Therefore, the LCO for isolating unborated water sources in MODE 6 is not required. This is an ITS Category (i) change.

106. ITS 3.9.3

3.9Q22 Review of the Actions Bases is pending response to LCO Q's.

Response: See responses to 3.9Q24, 3.9Q25, 3.9Q26, 3.9Q27, and 3.9Q29

3.9Q23 Refer to Bases markup for identification of editorial comments

Response: To be discussed at the next meeting.

- i. Incorporation of approved Traveller CEOG-02, C.3. This traveller was revised to delete the word "required" since there are only two independent source range channels. This is an ITS Category (iv) change.
- ii. Incorporation of approved Traveller BWOG-03, C.6.
- iii. The Completion Time for Required Action B.2 was revised to delete the 4 hour verification of the boron concentration limit since Ginna Station currently does not have this requirement. This requirement is performed when two source range neutron flux channels become inoperable. At the point in time when two source range channels become inoperable, the refueling boron concentration is assumed to be within limit. Verification that the boron concentration is within limit had been previously demonstrated by the periodic performance of SR 3.9.1.1. Since CORE ALTERATIONS and the addition of positive reactivity have been suspended (as a result of one inoperable source range channel), core reactivity conditions will remain stable. Therefore, the need to perform an additional verification within 4 hours is not necessary. Confirmation that core reactivity remains stable will continue to be performed every 12 hours. This is an ITS Category (iii) change.

3.9Q24 The intent of the 4 hour completion time to perform SR 3.9.1.1 is to confirm the assumption that the boron concentration is within required limits. WOG travelers are needed to start the generic change process. Is there an operational hardship associated with this SR?

Response: The initial 4 hour performance of SR 3.9.1.1 is being changed by a WOG traveller scheduled to go to the NRC by November 1. There are no significant operational hardships associated with this SR. However, as discussed above, the proposed change is not explicitly required by the CTS. Considering the water volume involved, and assuming boron concentration to be within the limit when the two source range channels become inoperable (as can be demonstrated by the previous performance of SR 3.9.1.1) and suspension of CORE ALTERATIONS and positive reactivity additions (as required by Actions A.1 and A.2) the core reactivity will be stable and any changes in boron concentration will occur very slowly. Therefore, the need to verify boron concentration within 4 hours is unnecessary. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #195.]

- iv. The Actions were revised to add a condition to address the loss of the audible count rate function. The audible count rate function is an initial assumption of the boron dilution during refueling event

at Ginna Station. Audible count rate is provided by one of the two required OPERABLE source range neutron flux channels. The addition of the Actions is consistent with current Ginna Station TS 3.8.1.c. This is an ITS Category (ii) change.

- v. The bases were revised as follows (these are ITS Category (iv) changes):
 - a. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with the LCO and other bases sections.

3.9Q25 Insert 3.9.7.b, page B 3.9-10

The proposed addition to the SR 3.9.2.1 appears to be an operator note for application of SR 3.0.1 and the use and application section of the TS. Explain how the added text elucidates the basis for the required TS surveillance.

Response: SR 3.0.1 states that all SRs must be met for the LCO to be considered met. However, SRs are not required to be performed on equipment which has been declared inoperable. In MODE 6, a CHANNEL CHECK is required every 12 hours. If one source range is declared inoperable, a CHANNEL CHECK is not required for this source range. However, a CHANNEL CHECK is still required for the second source range or it would then have to be declared inoperable per SR 3.0.1. Since there is no other neutron detector available in this instance, the CHANNEL CHECK requirements must be capable of being met by other means. The definition of CHANNEL CHECK in ITS Section 1.1 states that CHANNEL CHECKS "shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter." However, there is no further explanation of what to do when this is not possible. Thus, Insert 3.9.7.b was added. RG&E agrees that Insert 3.9.7.b is somewhat misleading and proposes to revise this as follows:

"If one channel is inoperable, a CHANNEL CHECK of the operable channel can consist of ensuring consistency with known core conditions since the Required Actions for the inoperable channel requires the suspension of CORE ALTERATIONS and positive reactivity addition."

Comment #111 has been opened to address this. A review of all other ITS required CHANNEL CHECKS indicates that an immediate shutdown must be entered if the number of channels falls below two (i.e., there is at least 2 channels which can be compared at all times) except as follows: (1) intermediate range channels; (2) source range channels; and (3) radiation monitors used for leakage detection. For the intermediate range channels, with one channel inoperable, the redundant channel can be compared to the source range channels such that a CHANNEL CHECK can still be made. For the source range channels below MODE 2, there is no other channel which could be checked such that bases clarifications are required to SR 3.3.1.1. For the radiation monitors, ITS SR 3.4.15.1 does not provide sufficient information since there is only one channel required to

be OPERABLE by ITS LCO 3.4.15. These will also be addressed by Comment #111. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #195.]

- b. The text was revised to provide consistency with the LCO addressing the audible count rate function supplied from either of the OPERABLE source range channels.
 - c. Ginna Station was designed and built prior to the issuance of the GDC contained in 10 CFR 50, Appendix A. However, the draft GDC issued by the Atomic Industrial Forum (AIF) in 1967 were utilized in the design of Ginna Station. The bases were revised to reflect this difference.
 - d. The text was revised to clarify that normal cool-down of the coolant volume for the purposes of temperature control is not considered as an addition of positive reactivity.
- vi. Various wording changes were made to improve the readability and to reflect plant-specific nomenclature (e.g., replace "monitors" with "channels"). This is an ITS Category (iv) change.

3.9Q26 Provide a design basis reason for replacing CTS requirements for monitor operability with ITS requirements for channel operability.

Response: The wording change was made to be consistent with plant-specific nomenclature. The LCO bases state that "to be OPERABLE, each channel must provide visual indication and at least one of the two channels must provide an audible count rate function in the control room." Relocation of this level of detail to the bases is consistent with the rest of the ITS and NUREG-1431.

3.9Q27 Delete the proposed reference to source range instrumentation in SR 3.9.2.2. This change is unnecessary to establish appropriate LCO SRs. Refer to Section 3.3 format and SR 3.0.1.

Response: RG&E agrees to delete this reference in the SR. Comment #110 has been opened to address this.

3.9Q28 Acceptance of the 24 month channel calibration frequency is pending staff review.

Response: No response required at this time.

- vii. Two Required Actions, similar to those for Condition A with one inoperable source range flux channel, were added to Condition B to require immediate suspension of CORE ALTERATIONS and positive reactivity additions. These changes provide a human factors improvement since Required Actions A.1, A.2, and B.1 all have immediate Completion Times. Locating all these requirements into Condition B is easier for operations personnel to implement. This is an ITS Category (iii) change.

3.9Q29 WOG travelers are needed to start the generic change process.

Response: The WOG rejected this change since it is not a technical issue. Suggest this be discussed during Ginna site visit.

107. ITS 3.9.4

3.9Q30 Provide plant specific LCO and Bases markup consistent with current TS requirements. Dose consequence calculations assume TS exist for personnel and equipment doors.

Response: See response to question 3.9Q7.

- i. LCO 3.9.4, "Containment Penetrations" and associated Bases, as presented in NUREG-1431, were not added and subsequent LCOs and Bases have been renumbered in consecutive order. Ginna Station has a plant specific safety analysis (UFSAR Section 15.7.3.3) for a fuel handling accident inside containment which assumes no isolation of the containment and no filtration following the accident. The NRC has concluded that this analysis is "well within" 10 CFR 100 limits (Ref. 3.9.2). Since LCO 3.9.4 requirements only ensure fission product radioactivity release from containment due to a fuel handling accident during refueling are "well within" 10 CFR 100 limits (see Bases), these requirements were not added. This change is discussed in detail in section C.2 of this Attachment (item 18.i). As a result of the deletion of ITS 3.9.4, approved Travellers BWOG-03, C.3, BWOG-03, C.6, WOG-05, C.2, and WOG-05, C.3, were not incorporated. This is an ITS Category (i) change.

108. ITS 3.9.5

3.9Q31 Refer to Bases markup for identification of editorial comments

Response: To be discussed at the meeting.

- i. Incorporation of approved Traveller BWOG-03, C.2.
- ii. SR 3.9.5.1 was revised to remove the flow rate for the RHR loop in operation. For Ginna Station, the boron dilution event is the only event postulated to occur in MODE 6 which assumes the RHR system in operation. The Ginna Station safety analysis for boron dilution in MODE 6 (UFSAR Section 15.4.4.2) assumes uniform mixing of the borated coolant as a result of a RHR pump being in operation and does not specify a given flow rate. Therefore, there is no analytical basis for the inclusion of a flow rate in the SR. The words "and circulating reactor coolant" were also deleted and relocated to the bases. This is an implied function for an RHR loop in operation and is consistent with the safety analysis and SR 3.4.8.1. This is an ITS Category (i) change.

3.9Q32 The SR 3.9.5.1 Bases do not confirm the justification that RHR pump flow rate and reactor coolant circulation are not required to verify LCO compliance. Discuss this discrepancy, include discussion of the criteria for establishing pump operability. Generic changes to the NUREG require WOG travelers to start the change process.

Response: A WOG Traveller has been initiated to change SR 3.9.5.1 to state "verify one RHR loop is OPERABLE and in operation." The justification for this traveller is that plants which do not credit isolation of the RCS to prevent boron dilution events only credit the use of RHR to ensure mixing of the borated coolant in MODE 6. No specific flow rate is specified in the accident analysis for this. Comment #112 has been opened to revise the proposed SR 3.9.5.1 to reflect this traveller and to track its status. With respect to relocation of "and circulating reactor coolant" to the bases, this was not done as stated in Change 108.ii. Comment #113 has been opened to delete the associated two sentences since this relocation is not required based on the wording of the proposed SR bases.

- iii. Incorporation of approved Traveller WOG-05, C.4.
- iv. Required Action A.4 and associated bases were not added since Ginna Station TS currently do not contain this requirement. The ITS Required Action to "close all containment penetrations providing direct access from containment atmosphere to outside atmosphere" is based on the scenario that there are no RHR loops in operation. This could lead over a period of time to boiling of the coolant and, should water level not be maintained, eventually challenge the integrity of the fuel cladding, which is a fission product barrier. The closure of "all containment penetrations" is only provided to limit the release of radioactive gases to the atmosphere. Plant procedures and administrative controls were established at Ginna Station in response to Generic Letter 88-017 (Ref. 3.9.3). These procedures and administrative controls include (1) providing at least two adequate means of adding inventory to the RCS and (2) closing containment penetrations during reduced RCS inventory operation. These procedures and administrative controls were verified by the NRC to be adequately implemented (Ref. 3.9.4). Since previously approved controls are in-place to ensure radioactivity releases from challenges to the integrity of the fuel cladding, adding additional restrictions at a higher water level are not necessary. This is an ITS Category (i) change.
- v. The bases were revised as follows (these are ITS Category (iv) changes):
 - a. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other bases sections.
 - b. The bases was revised since there is no explicit analysis assumptions for the decay heat removal function of the RHR System in MODE 6. There is, however, an assumption in the boron dilution event that one RHR pump is in operation.
 - c. The bases was revised to clarify that while ultimately it may be possible to achieve criticality without the presence of mixing (e.g., thermal or boron stratification), the boron dilution analysis assumes the coolant remains a homogeneous mixture. Additionally, the bases was revised to eliminate the



temperature indicating device since the definition of operability for the RHR loop would encompass this device.

- vi. Various wording changes were made to improve the readability and understanding of the Required Action consistent with LCO 3.4.8.
- vii. The title for LCO 3.9.5 was revised to be "RHR and Coolant Circulation - Water Level \geq 23 ft" which is consistent with the Applicability, Ginna Station procedures, and all activities which relate to shutdown operations. This is an ITS Category (iv) change.

109. ITS 3.9.6

3.9Q33 Refer to Bases markup for identification of editorial comments.

Response: To be discussed at the next meeting.

- i. Incorporation of approved Traveller BWOG-03, C.6.
- ii. Incorporation of approved Traveller BWOG-03, C.2. This traveller was modified to provide various wording changes to improve the readability and understanding of the bases. This is an ITS Category (iv) change.
- iii. Required Action B.3 and associated bases were not added since Ginna Station TS currently do not contain this requirement. The ITS Required Action to "close all containment penetrations providing direct access from containment atmosphere to outside atmosphere" is based on the scenario that there are no RHR loops in operation. This could lead over a period of time to boiling of the coolant and, should water level not be maintained, eventually challenge the integrity of the fuel cladding, which is a fission product barrier. The closure of "all containment penetrations" is only provided to limit the release of radioactive gases to the atmosphere. Plant procedures and administrative controls were established at Ginna Station in response to Generic Letter 88-017 (Ref. 3.9.3). These procedures and administrative controls include (1) providing at least two adequate means of adding inventory to the RCS and (2) closing containment penetrations during reduced RCS inventory operation. These procedures and administrative controls were verified by the NRC to be adequately implemented (Ref. 3.9.4). Since previously approved controls are in-place to ensure radioactivity releases from challenges to the integrity of the fuel cladding, adding additional restrictions at a higher water level are not necessary. This is an ITS Category (i) change.
- iv. SR 3.9.6.1 was revised to remove the flow rate for the RHR loop in operation. For Ginna Station, the boron dilution event is the only event postulated to occur in MODE 6 which assumes the RHR system in operation. The Ginna Station safety analysis for boron dilution in MODE 6 (UFSAR Section 15.4.4.2) assumes uniform mixing of the borated coolant as a result of a RHR pump being in operation and does not specify a given flow rate. Therefore, there is no analytical basis for the inclusion of a flow rate in the SR. The words "and circulating reactor coolant" were also deleted and

relocated to the bases. This is an implied function for an RHR loop in operation and is consistent with the safety analysis and SR 3.4.8.1. This is an ITS Category (i) change.

3.9Q34 The SR 3.9.6.1 Bases do not confirm the justification that RHR pump flow rate and reactor coolant circulation are not required to verify LCO compliance. Discuss this discrepancy, include discussion of the criteria for establishing pump operability. Generic changes to the NUREG require WOG travelers to start the change process.

Response: The response to this question is addressed in the response to question 3.9Q32.

- v. The bases were revised as follows (these are ITS Category (iv) changes):
 - a. Various wording changes were made to improve the readability and understanding of the bases. This includes providing consistency with other bases sections.
 - b. The bases was revised since there is no explicit analysis assumptions for the decay heat removal function of the RHR System in MODE 6. There is, however, an assumption in the boron dilution event that one RHR pump is in operation.
 - c. The bases was revised to clarify that while ultimately it may be possible to achieve criticality without the presence of mixing (e.g., thermal or boron stratification), the boron dilution analysis assumes the coolant remains a homogeneous mixture. Additionally, the bases was revised to eliminate the temperature indicating device since the definition of operability for the RHR loop would encompass this device.
- vi. Various wording changes were made to improve the readability and understanding of the Conditions and Required Actions consistent with LCO 3.4.8.

3.9Q35 When current refueling TS 3.8.1.d, 3.8.1.g are not met TS action 3.8.2 requires refueling work to cease, repairs to be initiated, and prohibits any operations that may increase the reactivity of the core. TS 3.8.3 further requires that if no loop is in operation then in addition to TS 3.8.2 the shutdown purge and minipurge penetrations are to be isolated within 4 hours. Justify any less restrictive changes to these TS requirements that are proposed by the May submittal NUREG markup.

Response: CTS 3.8.2 is addressed by ITS LCOs 3.9.1 through 3.9.5. The Required Actions for these LCOs are equivalent to or more restrictive to CTS 3.8.2 (other than those proposed to be relocated). That is:

- a. The CTS 3.8.2 requirement that "refueling of the reactor shall cease" is equivalent to "suspending CORE ALTERATIONS" for ITS LCOs 3.9.1, 3.9.2, and 3.9.5 since CORE ALTERATIONS is defined

as "movement of any fuel ... within the reactor vessel." This requirement is also equivalent to "suspend loading of irradiate fuel assemblies in the core" for ITS LCOs 3.9.3. ITS LCO 3.9.4 does not have this requirement; however, LCO 3.9.5 prevents fuel movement with less than 23 ft of water above the top of the reactor vessel flange such that this does not have to be specified in LCO 3.9.4 which only applies with less than 23 ft of water.

- b. The CTS 3.8.2 requirement that "work shall be initiated to correct the violated conditions so that the specified limits are met" is not required to be stated in the new Required Actions since this is always an option for exiting the LCO in the ITS.
- c. The CTS 3.8.2 requirement that "no operations may increase the reactivity of the core shall be made" is equivalent to "suspend positive reactivity additions" for LCOs 3.9.1, 3.9.2, and 3.9.5. This requirement is also equivalent to "suspend all operations involving reduction in RCS boron concentration" for LCOs 3.9.3 and 3.9.4 since other than reducing boron concentration, the only other method to increase reactivity is a temperature change which is directly affected by the loss of the RHR pumps (i.e., a temperature increase can occur if the required number of pumps is reduced).

CTS 3.8.3 only applies if CTS 3.8.1.d is not met. In this instance, it requires isolation of the shutdown purge and mini-purge valves. The loss of all RHR when water is ≥ 23 ft above the reactor vessel flange will not result in boiling until hours, if not days, later. In fact, the proposed shutdown rule does not even address this issue due to the time available to plant operators before boiling is reached, let alone uncovering of the core with subsequent fuel damage. Therefore, this requirement should not be retained in the ITS. [This response was changed as a result of the 11/16/95 Appeal meeting. See comment #221.]

- vii. The title for LCO 3.9.5 was revised to be "RHR and Coolant Circulation - Water Level < 23 ft" which is consistent with the Applicability, Ginna Station procedures, and all activities which relate to shutdown operations. This is an ITS Category (iv) change.

110. ITS 3.9.7

3.9Q36 Refer to Bases markup for identification of editorial comments and requests for technical clarifications

Response: To be discussed at the next meeting.

- i. Incorporation of approved Traveller BWOG-03, C.6.
- ii. The bases were revised as follows (these are ITS Category (iv) changes):
 - a. Various wording changes were made to improve the readability

and understanding of the bases. This includes providing consistency with other bases sections.

3.9Q37 Where is the 100 hour fuel decay prior to fuel handling limit in the ITS?

Response: The 100 hour fuel decay limit is not in the ITS LCOs. The relocation of this requirement to the Bases is consistent with the NRC Staff position to relocate (in accordance with the Criteria) the Standard Technical Specification for "Decay Time."

- b. The bases was revised to clarify that the fuel handling accident inside containment is "well within" the exposure guidelines of 10 CFR 100. This fractional release limit has been previously approved by the NRC (Ref. 3.9.2).
- iii. The Applicability was revised to relocate the "during movement of irradiated fuel assemblies within containment" prior to "during CORE ALTERATIONS." The existing Applicability is very confusing with the exception to CORE ALTERATIONS provided. This human factors improvement is preferred by licensed personnel. This is an ITS Category (iv) change.

Section 4.0 Current TS

44. Technical Specification 5.1

- i. TS 5.1.1, TS 5.1.2, and Figure 5.1-1 The description and figure of the site area boundary and exclusion area boundary was not added to the new specifications consistent with Traveller CE0G-03, C.1. Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR, Section 2.1.2). This is a Ginna TS Category (iii) change.

4.0Q1 Improved TS justification 111.i states that traveler CE0G-03, C.1 was not incorporated. Explain the apparent conflict between justifications 44.i and 111.i.

Response: The ITS justification 111.i should have stated that Traveller CE0G-03, C.1 was incorporated and was revised to reflect the Ginna CTS description. This is consistent with the incorporation of this traveller into NUREG-1431, Revision 1 which denotes, in brackets, the text description of the site location. The necessary change to ITS justification 111.i is provided under the ITS review section.

4.0Q2 Provide a discussion of change to justify deleting the site ownership statement in existing TS 5.1 and relocating the statement that [the accident analysis] assumptions are that the unrestricted area boundary coincides with the exclusion area boundary for evaluating radiological TS. The applicable regulation for design features TS is 10 CFR 50.36(c)(4).

Response: The site ownership description in the CTS 5.1 is not being deleted

but relocated from the TS. The description of the site ownership and the requirements of TS 5.1.1 and 5.1.2 are relocated to the UFSAR Section 2.1.2 and Figure 2.1-2. No technical change to these requirements have been made. The change control process governing future changes is in accordance with 10 CFR 50.59.

- 4.0Q3 The staff must be able to conclude that the improved TS support "adequate protection of the public health and safety" for evaluating radiological consequences related to 10 CFR Part 100 limits. To achieve this change in the ITS, retain the existing description of the site exclusion area by adding a one sentence description of the site exclusion area boundary to the ITS. The staff accepted: "The exclusion area boundary shall have a radius of _____ from the center line of the reactor" for recently issued BWR/6 plants conversion TS.

Response: To be discussed at the meeting.

45. Technical Specification 5.2

- i. TS 5.2 - The description of the containment design features was not added. Specific containment features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR Sections 3.8.1 and 6.2). This is a Ginna TS Category (iii) change.

- 4.0Q4 Specify the document each relocated item will be moved to and state the change control process governing future changes to that document.

Response: The containment design features as described in TS 5.2 are relocated to the UFSAR. This is consistent with the information required by 10 CFR 50.36 as contained in NUREG-1431. The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

46. Technical Specification 5.3

- i. TS 5.3.1.a and TS 5.3.1.c - The description of the reactor core design features was revised consistent with the standard guideline of NUREG-1431. The section now includes the amount, kind, and source of nuclear material related to the reactor core. This is a Ginna TS Category (v.c) change.

- 4.0Q5 Provide a discussion of change to justify deleting: (1) the approximate total mass of uranium dioxide pellets, (2) the use of Zircaloy-4, (3) the descriptions of the fuel rod positions in a fuel assembly, (4) reporting requirements regarding fuel assembly rod replacement limits per refueling, (5) fuel assembly design features regarding guide tubes, instrument thimbles, and array, and (6) RCC design parameters including control cladding consisting of silver-

indium-cadmium in the design features TS section.

- Response:
- (1) *The approximate total mass of the uranium dioxide pellets has been relocated to UFSAR Table 4.2-1.*
 - (2) *The use of Zircaloy-4 have been retained in ITS 4.2.1. This information is also provided in UFSAR Sections 4.2.3.1, 4.2.3.1.6, and 4.1.3.*
 - (3) *The descriptions of the fuel rod positions has been relocated to UFSAR 4.2.3.1.*
 - (4) *The reporting requirements for fuel rod replacement will be relocated to Ginna Station procedure A-25.6 (attached). Comment #166 has been opened to address this issue.*
 - (5) *The fuel assembly design features have been relocated to UFSAR Section 4.2.3.1 and Table 4.2-2.*
 - (6) *The RCC design parameters have been relocated to UFSAR 4.1.1, 4.2.3.1, 4.2.3.2, and 4.2.2.*

The change control process governing future changes to the UFSAR and procedure A-25.6 is in accordance with 10 CFR 50.59. These relocated items do not meet the criteria for inclusion in the Design Features of technical specifications per 10 CFR 50.36.

- ii. TS 5.3.1.b - The description of the fuel storage design feature with respect to the maximum enrichment weight percent was revised and relocated to new Specification 4.3.1. The changes are in accordance with the changes discussed in item 47.ii, below. These are Ginna TS Category (v.c) and (i) changes, respectively.

4.0Q6 Provide a discussion of change to justify deleting the enrichment reload discussion in existing TS 5.3.1.b. Further, explain how the existing TS 5.3.1.b reload fuel enrichment discussion is related to ITS 4.3.1.2.a discussion on design features for new fuel storage racks.

Response: *The CTS 5.3.1.b denotes the requirements for new fuel similar to the requirements denoted in ITS 4.3.1.2.a. The only difference between the CTS and ITS is that the ITS is related to storage of new fuel while the CTS can be interpreted to relate to acceptance of reload fuel. A fuel handling accident of the reload fuel prior to placement in the storage racks is not of concern with respect to offsite doses since the fuel has not been irradiated with source terms being generated. Therefore, relating CTS 5.3.1.b requirements to storage of the new fuel per ITS 4.3.1.2.a is acceptable.*

The discussion of the fuel delivered prior to and after January 1, 1984 can be deleted since this information is encompassed in the ITS requirement of a "maximum" enrichment and is therefore no longer applicable. That is, the fuel handling accidents for the new fuel pool (with respect to criticality concerns) have been performed assuming that the fuel pool is arranged with fuel of the maximum

enrichment limit. Also, the information provided is consistent with that contained in NUREG-1431.

iii. TS 5.3.2 - The description of the reactor coolant system (RCS) design features was not added. Specific RCS features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR Section 3.7.1 and Chapter 5). This is a Ginna TS Category (iii) change.

iv. TS 5.3.1.b - This was revised to increase the fuel enrichment limit from 4.25 weight percent to 5.05 weight percent. This change has been evaluated and found to be acceptable with respect to postulated fuel handling accidents (Ref. 29). This is a Ginna TS Category (v.b.46) change.

4.0Q7 Implementation of this proposed change requires a staff SER for the Westinghouse criticality analysis (Ref. 29). Provide a reference to the staff SE.

Response: The NRC staff approved this change via letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: "Safety Evaluation of Rochester Gas and Electric's Proposed Criticality Analysis of the Ginna New and Spent Fuel Racks / Consolidated Rod Storage Canisters (TAC No. M92188)," dated August 30, 1995.

47. Technical Specification 5.4

i. TS 5.4.1, 5.4.2, 5.4.6, and Figures 5.4-1 and 5.4-2 - The description of the fuel storage design features denoting spent fuel storage regions and borated water concentrations was relocated to Chapters 3.7 and 3.9. These features are discussed in LCO, 3.7.11, LCO 3.7.12, LCO 3.7.13, and LCO 3.9.1 as appropriate. In addition, appropriate Required Actions were added in the event that SFP water level, boron concentration, or SFP region storage requirements are not met. This is a Ginna TS Category (i) change.

ii. TS 5.4.2-- The description of the fuel storage design features was revised. The revision to these features are based on a revised criticality analysis supporting the proposed 18 month fuel cycle (Reference 29). The description of these features follow the standard guideline of NUREG-1431 which would include the amount, kind, and source of special nuclear material with the exception that nominal center to center spacing between the fuel assemblies was not added. This is a Ginna TS Category (v.c) change.

4.0Q8 Provide a discussion of change to justify deleting (1) that fuel is stored vertically, (2) fuel enrichments for unirradiated fuel assemblies delivered prior to January 1, 1984, and (3) references to unirradiated fuel assemblies delivered after January 1, 1984.

Response: (1) The fact that fuel is to be stored vertically is provided in

UFSAR Section 9.1.2.2.2. The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

(2) *See response to 4.0Q6. Also, the fuel storage limitations are now controlled by ITS LCO 3.7.13.*

(3) *See response to 4.0Q6. Also, the fuel storage limitations are now controlled by ITS LCO 3.7.13.*

4.0Q9 Implementation of proposed changes to fuel enrichments requires a staff SER for the Westinghouse criticality analysis (Ref. 29). Provide a reference to the staff SE.

Response: See response to 4.0Q7.

iii. TS 5.4.3 - The description of the fuel storage design feature denoting the 60-day limit on storage of discharged fuel assemblies in Region 2 was not added. No screening criteria applies for the time limit on storage of discharged fuel assemblies in Region 2. The current 60-day limit was established to provide sufficient margin in spent fuel pool temperature calculations as a result of decay heat loads in Region 2 from discharged fuel assemblies (Reference 39). Although the spent fuel pool cooling system and, thus, the associated restriction on heat load prevent structural integrity damage to the spent fuel pool, they are not assumed to function to mitigate the consequences of a design basis accident (DBA). The restriction on heat load is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The restriction on heat load is a non-significant risk contributor to core damage frequency and offsite doses. Since no NRC Final Policy Statement technical specification screening criteria apply, this requirement is relocated to the TRM. This is a Ginna TS Category (iii) change.

4.0Q10 All proposed TS 5.4.3 changes were not discussed in justification 47.iii. Provide a markup of all proposed changes with appropriate justification.

Response: (1) The discussion related to storage in a close packed array utilizing fixed neutron poisons in each location has been relocated to UFSAR Sections 9.1.2.1.12 and 9.1.2.2.2. The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

(2) The 60 day decay time is discussed in justification 47.iii above.

(3) The average burnup and enrichment limits ensuring that $k_{eff} \leq 0.95$ discussion has been relocated to ITS 3.7.13 and associated Figure 3.7.13-1.

iv. TS 5.4.4 and 5.4.5 - These were revised consistent with References 29 and 39 to provide the amount, kind, and source of material which is stored in the canisters. TS 5.4.4.b and 5.4.5 were not added the

new specifications for the reasons discussed in item 47.iii above. These are Ginna TS Category (v.c) and (iii), respectively.

4.0Q11 Show how each commitment in TS 5.4.5 and TS 5.4.4.b will be maintained in the improved TS 4.3.1.1. Explain why proposed changes discussed in this justification are administrative changes.

Response: (1) The requirements of CTS 5.4.4.a have either been retained in ITS 4.3.1.1.c (last sentence) or relocated to the UFSAR as discussed in the responses to 4.0Q8 and 4.0Q10.

(2) The requirements of CTS 5.4.4.b only relate to the decay heat removal requirements which have not been retained as discussed in the justification 47.iii. This decay heat removal requirement is relocated to UFSAR Section 9.1.3 (Section 9.1.3.4.1.6 for the consolidated fuel canisters). The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

(3) The requirements of CTS 5.4.4.5 allow the canister stored in Region RGAF2 to exceed the requirements of CTS 5.4.2 and 5.4.3. This allowance should have been retained since this canister will not meet the requirements of ITS LCO 3.7.13 as required by ITS 4.3.1.1.c. Comment #167 has been opened to add this CTS exception back into the ITS and to correct the reference in ITS 4.3.1.1.c to Specification 3.7.13 versus 3.7.17.

v. TS 5.4 - This was revised to include descriptions of the SFP drainage system and capacity. This information is currently contained in the bases for this section. Since NUREG-1431, Chapter 4 does not contain any bases, this information has been relocated to the specification. This is a Ginna TS Category (i) change.

4.0Q12 The proposed revisions are not clearly marked. Provide legible copies of current TS pages 5.4-2, 5.4-3 and 5.4-4 and identify and justify all proposed changes.

Response: These pages are Bases pages and are not required to be marked and justified since the Design Features section no longer has bases pages and is not required to by 10 CFR 50.36. The information provided by these bases is contained in UFSAR Section 9.1. The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59. Therefore, the requested pages have been provided even though the information contained on these pages is being relocated.

48. Technical Specification 5.5

i. TS 5.5 - The description of the waste treatment systems design features was not added. No screening criteria apply for the description of these features. Specific waste treatment systems features are either covered in the Technical Specification LCO's or have been relocated to other licensee controlled documents and, therefore, do not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Since the description of these design features

does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR Chapter 11). This is a Ginna TS Category (iii) change.

Current 4.0 TS

111. ITS 4.1

- i. Incorporation of approved Traveller CEOG-03, C.1 (Rev.1). Additional changes were also made to reflect Ginna Station design issues. These are ITS Category (iv) changes.
- ii. A typographical or minor clarification is identified. This is an ITS Category (iii) change.

112. ITS 4.2

- i. Approved Traveller NRC-01, C.1 was not incorporated since this does not apply to Ginna Station.

4.0Q13 Explain why the traveler does not apply to Ginna.

Response: The Traveller proposed to revise the phrase "zirconium alloy" with "Zircalloy or ZIRLO." This Traveller change is misleading and inconsistent with the nomenclature used at Ginna in describing this feature. The phrase "zirconium clad" more accurately represents the description of this feature (see UFSAR Table 4.2-3 which shows that the current fuel [last column] is clad by both zircaloy and inconel). [This response was changed as a result of meetings the week of 11/20/95. See comment #226.]

- ii. The control rod assembly material description was not added to the specifications since the specification already requires NRC approval of this material. Therefore, any change in material would require both a TS change and NRC approval. This is a ITS Category (i) change.

4.0Q14 These generic changes require a staff approved traveler.

Response: The phrase as denoted on the NUREG is misleading in that it creates two compliance issues. The first compliance is the specific control material that is identified. The second compliance is the phrase "as approved by the NRC." Additionally, the structure of the bracketed phrase is confusing in that it provides, with the logical connector "or," a case where all of the control materials can be listed and only those "approved by the NRC" are required to be met. Lets discuss next week.

113. ITS 4.3

- i. ITS.4.3.1.1.c and 4.3.1.1.d were not added to the new specifications since these items are not currently in the Ginna Station Technical Specifications. The spacing of fuel assemblies in the SFP is ultimately controlled by ITS 4.3.1.1.a and 4.3.1.1.b which state

that the spent fuel storage racks are designed to limit $k_{eff} \leq 0.95$ assuming the pool contains fuel assemblies with 5.05 weight percent U-235 enrichment and is fully flooded with unborated water. This is an ITS Category (i) change.

4.0Q15 Explain the design or regulatory burden basis for not wanting to upgrade ITS to these design parameters?

Response: The addition of this information is not necessary to meet compliance with 10 CFR 50.36 and is not in the current TS. While the spacing of the fuel assemblies in the spent fuel pool is an assumption of the spent fuel pool criticality analysis, ITS 4.3.1.1.b requires that the pool be maintained with a $k_{eff} \leq 0.95$ assuming that it is fully flooded with unborated water. Consequently, the ultimate criteria for the design and storage of fuel in the spent fuel pool is to maintain k_{eff} within limits. This is being retained. The spacing of the fuel assemblies is discussed in the UFSAR (Table 9.1-1). The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

ii. ITS 4.3.1.1.e and 4.3.1.1.f were not added to the new specifications since these items are controlled by new ITS 3.7.17. The revised ITS 3.7.17 provides limits on the storage of fuel in the two SFP regions with appropriate action statements such that TS 4.3.1.e and 4.3.1.f are no longer required. Consequently, approved Traveller NRC-06, C.1 (Rev. 1) was also not added. This is an ITS Category (i) change.

4.0Q16 Give the design or regulatory burden basis for not wanting to upgrade ITS to these design parameters?

Response: Since these requirements are controlled by new ITS 3.7.17, the addition of this information to Section 4.0 would lead to the potential for conflicting information or for mis-interpretation of requirements. The ITS attempts to eliminate duplicate requirements for this reason by relocating these operational details to the ITS Chapter 3.7. Comment #168 has been generated to create and track a traveller on this issue.

[CTS 113.iii-M]
iii. The use of consolidated rod storage canisters was added to the new specifications consistent with current Ginna Station TS 5.4.4 and References 29 and 39.

4.0Q17 Many of the details from TS 5.4.4 were not incorporated into insert 4.3.1.c. Other TS requirements not in TS 5.4.4 were included in insert 4.3.1.c. Provide a more detailed markup of TS 5.4.4 showing how the current license requirements are proposed to be changed upon implementation of improved TS and explain the basis for each change.

Response: See response to 4.0Q11. The necessary remarkup of this CTS is attached.

4.0Q18 Give the basis for including canister operability requirements

(i.e., insert 4.3.1.c reference to specification 3.7.17) in design features TS.

Response: This is consistent with the level of detail required for 4.3.1.1 by 10 CFR 50.36. [This response was changed as a result of meetings the week of 11/20/95. See comment #226.]

4.0Q19 Reference 39 increases spent fuel storage capacity to 1016 assemblies from 595. How is this related to the justification for canister operability limits?

Response: The reference to Reference 39 is an error. Instead, the following document should have been referenced (attached):

Letter from G.E. Lear, NRC, to R.W. Kober, RG&E, Subject: "Storage of Consolidated Fuel," dated December 16, 1985.

iv. ITS 4.3.1.2.d was not added to the new specifications since this is not currently in the Ginna Station Technical Specifications. The spacing of fuel assemblies in the new fuel pool is ultimately controlled by ITS 4.3.1.2.a and 4.3.1.2.b which state that the new fuel storage racks are designed to limit $k_{eff} \leq 0.95$ assuming the pool contains fuel assemblies with 5.05 weight percent U-235 enrichment and is fully flooded with unborated water. This is an ITS Category (i) change.

4.0Q20 Give the design or regulatory burden basis for not wanting to upgrade ITS to these design parameters?

Response: The addition of this information is not necessary to meet compliance with 10 CFR 50.36 and is not in the current TS. While the spacing of the fuel assemblies in the new fuel storage racks is an assumption of the new fuel criticality analyses, ITS 4.3.1.2.b requires that the pool be maintained with a $k_{eff} \leq 0.95$ assuming that it is fully flooded with unborated water. Consequently, the ultimate criteria for design and storage of fuel in the new fuel storage racks is to maintain k_{eff} within limits. This is being retained. The spacing of the fuel assemblies is discussed in the UFSAR (Table 9.1-1). The change control process governing future changes to the UFSAR is in accordance with 10 CFR 50.59.

v. The description of the new fuel storage racks was clarified to state that the description applied to the "dry" racks consistent with Reference 29. This is an ITS Category (iv) change.

4.0Q21 The markup provides the terminology "new fuel storage dry racks." Is this correct?

Response: The phrase "dry racks" more accurately represents the description of this feature. This is a plant specific nomenclature and was added to prevent mis-interpretation of the requirements. At Ginna Station, the fuel is received and stored in "dry racks" located in the Aux Building near the spent fuel pool. To be transferred to

containment, the fuel is moved to the spent fuel pool and then transferred by the gate transfer system.

- vi. ITS 4.3.2 was revised to reflect Ginna Station nomenclature and to provide clarification. These are ITS Category (iv) changes.

4.0Q22 How is "<23 feet above the top of the fuel assemblies" verified by plant procedures to be met?

Response: This is a new TS requirement for Ginna Station. The verification during refueling activities is addressed by ITS SR 3.7.11.1. Verification at other times is not required by CTS, nor by the ITS. [This response was changed as a result of meetings the week of 11/20/95. See comment #226.]

- vii. ITS 4.3.3 was revised to reflect Ginna Station nomenclature. This is a ITS Category (iv) change. It is noted that the value specified for the SFP capacity is a theoretical value based on the exclusive use of consolidated fuel canisters (i.e., no fuel assemblies). The current SFP is limited to 1016 fuel assemblies due to limitations of the SFP Cooling System (see the bases for current Ginna Station TS 5.4). Since the heat removal capability of the SFP Cooling System does not meet the criteria of the NRC Policy Statement, the new specification was based on the theoretical value.

4.0Q23 In the current TS the spent fuel capacities are identified specifically for region I and region II. Provide a basis for changing this designation.

Response: The CTS spent fuel capabilities are provided in Figure 5.4-1. This figure has been relocated to ITS Bases Figure B 3.7.13-1 which is under the Bases Control Program. The capacity limits found on this figure relate only to spent fuel pool cooling requirements which have not been added as discussed above in justification 47.iii. Therefore, the ITS 4.3.3 description was marked to be the theoretical capacity of the pool consistent with the bases found on CTS page 5.4-4.

Section 6.0 Current TS

1.xiii. TS 1.13 - The definition for Offsite Dose Calculation Manual (ODCM) was not added to the new specifications since it is no longer required. [L] The ODCM is described in new Specification 5.5.1. This is a Ginna TS Category (v.c) change.

15.viii. 3.5.5 *Radioactive Effluent Monitoring Instrumentation [RETS]*

Existing TS 3.5.5, and Table 3.5-5 *procedural requirements addressing limiting conditions for operation, their applicability, remedial actions, and reporting requirements* for establishing radioactive effluent monitoring instrumentation operability to ensure that the limits of existing TS 3.9.1.1 - liquid effluent concentrations in the circulating water discharge and existing TS 3.9.2.1 - gaseous wastes instantaneous dose rates as calculated in

the ODCM are not exceeded are not assumed in the accident analysis and are relocated to ODCM and the Effluent Controls Program described in improved TSs 5.5.1 and 5.5.4. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively.

6.0Q1 For current TS requirements show which programmatic controls and procedural details of TS 3.5.5 and Table 3.5-5 are included in the proposed ODCM Program (specification 5.5.1) and Radioactive Effluents Controls Program TS (specification 5.5.4) and which of these TS will be maintained outside the TS. Confirm that all NRC guidance in Generic Letter 89-01 for removal of these TS to administrative control TS programs has been met.

Response: CTS 3.5.5, 3.9, and 3.16 contain details related to implementation of the ODCM and RETS programs. All details contained in these CTS sections are proposed to be relocated to station procedures "as is" (i.e., no changes will be made other than minor editorial changes for additional clarity). Only the general requirements which implement the details of these CTS requirements will be retained in the ITS ODCM and Radioactive Effluents Controls Program specifications. The conformance with respect to GL 89-01 is addressed in the response to 3.9Q2.

19. Technical Specification 3.9 - Plant Effluents [RETS]

6.0Q2 For current TS requirements show which programmatic controls and procedural details of TS 3.9 are included in the proposed ODCM program and Radioactive Effluents Controls Program TS and which of these details will be maintained outside the TS. [R,T]

Response: See responses to 3.9Q2 and 6.0Q1.

- i. Existing TS 3.9.1.1 requirements for radioactive material released in liquid effluents of the circulating water discharge to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.

- ii. Existing TS 3.9.1.2 and existing TS 3.9.2.4 requirements for dose or dose commitment to individuals which results from cumulative liquid effluent discharges during normal operation over extended periods and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I, were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, radioactive liquid effluent dose projected value is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- iii. Existing TS 3.9.1.3 requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, GDC 60 and 10 CFR Part 50, Appendix I, Section II.D, were not added. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- iv. Existing TS 3.9.2.1 requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- v. Existing TS 3.9.2.2.a, TS 3.9.2.2.c, and TS 3.9.2.4 requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a

GINNA TS Category (iii) change.

- vi. Existing TS 3.9.2.2.b, TS 3.9.2.2.c, and TS 3.9.2.4 requirements for dose due to radioiodine, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days released with gaseous effluents were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, these gaseous effluents doses are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- vii. Existing TS 3.9.2.3 requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- viii. Existing TS 3.9.2.5 and TS 3.9.2.6 specific requirements for which limit concentration of oxygen in a gas decay tank and the quantity of radioactivity contained in each waste gas decay tank were not added. The level of detail is relocated to Explosive Gas and Storage Tank Radioactivity Monitoring Program described in new Specification 5.5.11 [R,T] and a more generic description is provided. This is a Ginna TS Category (iii) change.
- ix. Existing TS 3.9.2.7 requirements for the solid radwaste system which processes wet radioactive waste and operates in accordance with 10 CFR Part 50, Appendix A, for effluent control were not added. The operability of the system is not assumed in the analysis of any DBA or transient. Further, radioactive waste is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program [R,T] described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

23. Technical Specification 3.13 - *Snubbers*

- i. TS 3.13 - The requirements for snubbers operability were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to the Inservice Testing

Program described in new Specification 5.5.8 [R,T] and more generic program description is provided. This is a Ginna TS Category (iii) change.

6.0Q3 For current TS requirements show which programmatic controls of TS 3.13 are included in the inservice inspection and inservice test program TS and show which procedural details of TS 3.13 are proposed to be maintained in the improved TS, e.g., relocated to licensee-controlled documents?

Response: The issue of snubber OPERABILITY as contained in CTS 3.13 is addressed by the definition of OPERABLE - OPERABILITY in the ITS such that the associated system's LCO would be entered as applicable for an inoperable snubber. This is consistent with option 2 of CTS 3.13.2. With respect to snubber surveillance requirements (CTS 4.14), the current ISI Program for snubbers and the engineering documentation specifically referenced in this program (attached) contain the CTS requirements with the exception of certain text in CTS 4.14.1.d and Table 4.14-1 (e.g., the engineering documentation references Table 4.14-1 without duplicating it). This CTS text and table will be duplicated in the engineering documentation prior to ITS implementation such that CTS 4.14 will be relocated in its entirety to the ISI Program.

25.v. *LTOP Special Report*

Existing TS 3.15.1.3 contains a requirement to report use of the overpressure protection system to mitigate an RCS or RHR pressure transient in accordance with specification 6.9.2. The Special Report is required to include documentation of all challenges to the pressurizer power operated relief valves or RCS vent(s) and a description of the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any other corrective action. These requirements is detailed in improved TS 5.6.4, "Monthly Operating Reports" [T] and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. This is a Ginna TS Category (i) change.

26. Tech Spec 3.16 - *Radiological Environmental Monitoring [RETS]*

6.0Q4 For current TS requirements show which programmatic controls of TS 3.16 are included in the ODCM and Radioactive Effluents Controls Program TS and which details will be maintained outside TS.

Response: See responses to 3.9Q2 and 6.0Q1.

- i. Existing TS 3.16.1 and Table 3.16-1 for the radiological environmental program requires measurements of radiation and of radioactive materials in exposure pathways and measurement of specified radionuclides which lead to the highest potential radiation exposures for members of the public. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk



contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.

- ii. Existing TS 3.16.2 requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
 - iii. TS 3.16.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- 28.v.b. *Radioactive Effluent Monitoring Surveillances [RETS]*
Existing TS 4.1.4 and Table 4.1-5 radioactive effluent monitoring instrument functions required by this specification were not added to the new specifications since these process variables are not an initial condition or a DBA or transient analysis. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and were relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, [R,T] respectively. This is a Ginna TS Category (iii) change.
- 28.i.j. Table 4.1-1, Functional Units #18, #28, and #29 - The Surveillance requirements for radiation monitors R-1 through R-9 and R-17, emergency plan radiation instruments, and environmental monitors, were not added to the new specifications. These process variables are not an initial condition of a DBA or transient analysis. Therefore, the requirements specified for these functions do not

satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. [R] This is a Ginna TS Category (iii) change.

6.0Q5 For current TS requirements show which programmatic controls of TS 4.1.1 and Table 4.1-1 are included in the ODCM Program and Radioactive Effluents Controls Program TS and which details will be maintained outside TS.

Response: All requirements of CTS Table 4.1-1, Functional Units #18, #28, and #29 are proposed to be relocated to plant procedures which implement in the ITS ODCM and Radioactive Effluents Controls Programs. Essentially, the CTS requirements are being relocated as is to the new ITS programs with only minor editorial changes.

29. Technical Specification 4.2 Inservice Inspection

- i. Existing TS 4.2.1 requirements for the Inservice Inspection Program, which include Quality Groups A, B, and C components, high energy piping outside of containment, snubbers and steam generator tubes, were not added. The level of detail is relocated to licensee controlled documents (Ginna Station QA Manual, Appendix B) [R] and a more generic description is provided. This is a Ginna TS Category (iii) change.

6.0Q6 ISI program (TS 4.2) for Quality Groups A, B, and C components, high energy piping outside containment, snubbers and Steam Generator tube requirements are proposed to be moved to the IST program (improved TS 5.5.8) and SG tube surveillance program (improved TS 5.5.9). Also, inservice pump and valve testing contained in existing TS 4.2 are not discussed by change justification 29.i. For the current TS requirements show which programmatic and procedural details of the ISI program and the inservice pump and valve testing contained in existing TS 4.2 are to be maintained in improved TS. Discuss any proposed changes to the QA Manual Appendix C for the inservice pump and valve testing.

Response: Upon further review, RG&E has concluded that the proposed text added to NUREG 5.5.8 as change 120.xvii should be deleted. The proposed text relates to ISI program which is different from the IST discussed in this specification (see 10 CFR 50.55a (g) and (f), respectively). Therefore, the CTS 4.2.1 (i.e., 4.2.1.1, 4.2.1.2; 4.2.1.3, and 4.2.1.5) requirements for Quality Groups A, B, and C components and high energy piping is proposed to be relocated to the ISI program outside of TS (see attached-1 - Note that A, B, and C are equivalent to 1, 2, 3 in the ISI program). The CTS 4.2.1.4 acceptance criteria for SG tube inspections are retained in ITS 5.5.9. However, the SG tube inspection interval specified by CTS 4.2.1 are being relocated to the ISI program outside TS (see attached-2). With respect to snubbers, these are not really in the IST Program even though NUREG-1431, Rev. 1 states that they are. Instead, these are in the ISI Program (see response to 6.0Q5 and 10 CFR 50.55a(f) which only discusses pumps and valves with respect to



the IST Program). However, RG&E is willing to leave them in ITS 5.5.8 provided that their being located in the ISI program is acceptable. Finally, the IST requirements of CTS 4.2.1 and 4.2.1.6 are being relocated outside TS to the IST Program since this level of information is not in NUREG-1431, Revision 1. However, no changes to the current IST Program are required (see attached-3). Comment #151 has been opened to correct the NUREG markup to remove the reference to high energy piping and SG tubes in the IST program.

31. Technical Specification 4.4.3 RHR Systems Surveillances

- i. TS 4.4.4 - The requirements for the tendon stress surveillances were not added. The level of detail is relocated to the Pre-stressed Concrete Containment Tendon Surveillance Program described in new Specification 5.5.6 and a more generic program description is provided. [T, R] This is a Ginna TS Category (iii) change.

6.0Q7 Requirements proposed to be relocated need an evaluation comparing the proposed change to the criteria of 10 CFR 50:36 and identifying the location of the relocated requirement and the proposed control mechanism for future changes to the relocated requirement. Provide an evaluation of current TS requirements and show which programmatic and procedural details in TS 4.4.4 will be maintained in proposed TS 5.5.6. Existing TS 4.4.3 requirements for testing of the RHR system in the recirculation configuration were not added. The level of detail is relocated to the Primary Coolant Sources Outside Containment Program described in new Specification 5.5.2 and a more generic program description is provided. [T, R] This is a Ginna TS Category (iii) change.

Response: The details for the tendon stress surveillances in CTS 4.4.4 are being relocated to the Pre-Stressed Concrete Containment Tendon Surveillance Program (ITS 5.5.6). CTS 4.4.4 specifies the minimum tendon sample population, testing frequency, and acceptance criteria. ITS 5.5.6 requires that the "Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with a NRC approved program." NUREG-1431 references RG 1.35 is Specification 5.5.6 which specifies the minimum sample population, testing frequency, and acceptance criteria. Therefore, if the ITS 5.5.6 specifically referenced RG 1.35, there would be no technical change since the specification would reference a NRC approved methodology for sample population, testing frequency, and acceptance criteria instead of reiterating them in TS.

However, the RG&E program differs with respect to RG 1.35 with respect to several factors. Essentially, the RG&E program requires surveillances of a larger sample population and has more stringent acceptance criteria. Therefore, just referencing RG 1.35 would be a reduction in commitment. Instead, RG&E proposed to require "in accordance with a NRC approved program." The current program was approved by the NRC in a SER dated August 19, 1985 and NUREG-0821 (attached). As such, the program would contain all the specific details currently in TS, but would require NRC approval for changes to this program, similar to a technical specification change.

Therefore, there is no reduction in commitment and the criteria for inclusion within TS are not affected. However, RG&E would consider putting the NUREG words referencing RG 1.35 back into ITS 5.5.6 and controlling the additional requirements in procedure PT-27 (attached). [End]

6.0Q8 For the current TS requirements show which programmatic and procedural details in TS 4.4.3 will be maintained in improved TS 5.5.2.

Response: *CTS 4.4.3 contains specific criteria for testing pressure, acceptance criteria, correction actions, and test frequencies for systems that can contain containment sump fluid during the recirculation phase of an accident. This is being relocated to Primary Coolant Sources Outside Containment Program (ITS 5.5.2), which specifies the systems subject to this testing and requires "preventative maintenance and periodic visual inspections" and leak tests "at refueling cycle intervals or less." Therefore, the specific testing pressure, acceptance criteria, and corrective actions are being relocated to plant procedures which implement ITS 5.5.2. These procedures are under the control of procedure change process at Ginna Station which requires, as a minimum, a 10 CFR 50.59 screening. The testing frequency is also being changed from every 12 months to "refueling cycle intervals or less" in ITS 5.5.2. The ITS essentially define refueling cycle interval as 24 months (Note - this is not an ITS definition anywhere, should ITS 5.5.2 be changed to specify 24 months?). The increased surveillance interval from 12 months to 24 months is acceptable since the affected systems are normally filled with water. Leakage through these systems would be detected by operator walkdowns and during IST related pump tests which are performed quarterly. Therefore, this increased surveillance interval is considered acceptable.*

32.v. *Air Filtration System Surveillances*
TS 4.5.2.3 - The requirements denoting the Frequency and conditions of the air filtration system tests were not added to the new specifications. This level of detail is relocated to the Ventilation Filter Testing Program described in new Specification 5.5.10. In addition, the remaining requirements were all relocated to the Administrative Controls section. These are Ginna TS Category (iii) and (i) changes, respectively. [T, R]

6.0Q9 What licensee-controlled document will contain the frequency and condition requirements of the air filtration system tests proposed to be relocated? What mechanism will be used to control future changes to these requirements? Proposed specification 5.5.10 states that the test frequencies and methods will be performed, where practical, in accordance with Regulatory Guide 1.52. From this statement it is not possible to identify what changes, if any are being proposed for the existing TS programmatic and procedural details. For the current TS requirements show which programmatic and procedural details in TS 4.4.3 will be maintained in improved TS.

Response: The VFTP specifically references RG 1.52, Revision 2. This regulatory guide contains the same testing frequencies as specified in CTS 4.5.2.3.1, 4.5.2.3.2, 4.5.2.3.3, 4.5.2.3.4, 4.5.2.3.6, 4.5.2.3.7, and 4.5.2.3.8 (see sections 5.b and 5.c of RG 1.52). Therefore, these seven CTS sections are being relocated to ITS 5.5.10. The methyl iodide test requirements (CTS 4.5.2.3.1.c and 4.5.2.3.6.d) will be relocated to the station procedures. These procedures will be under the control of procedure change process at Ginna Station which requires, as a minimum, a 10 CFR 50.59 screening.

- vi. TS 4.5.2.3.6.a - These test requirements were revised to clarify that two separate tests are performed. A HEPA filter test and a charcoal adsorber bank test are separately performed with each requiring a limit of less than 3 inches of water. This is essentially equivalent to a combined test of less than 6 inches of water and is consistent with specified testing standards. This is a Ginna TS Category (vi) change.

6.0Q10 Explain what is meant by essentially equivalent? Explain the extent to which current procedures will need to change to accommodate the proposed change.

Response: CTS 4.5.2.3.6.a requires that the pressure drop across the combined HEPA filters and charcoal filters be less than 6 inches of water. ITS 5.5.10.c.1 and 5.5.10.c.3 limit the pressure drop to less than 3 inches for the HEPA and charcoal filters, respectively. Therefore, the total pressure drop remains less than 6 inches but the location of the pressure drop is limited (i.e., the HEPA filter cannot have a pressure drop of 4 inches and the charcoal filter 1 inch and continue to meet the ITS limits). The current testing procedures test the pressure drop across the HEPA and charcoal filters separately (see attached procedure PT-47.3, steps 6.5.3 and 6.6.4) such that only their acceptance criteria needs to be changed.

33. iv. *Diesel Fuel Oil Surveillances*
TS 4.6.1.d - The diesel fuel oil test requirements were relocated to new TS 5.5.12 and are proposed to be identified as a "program" consistent with the format of NUREG-1431. This is a Ginna TS Category, (i) change.

37. Tech Specs 4.10 *Radiological Environmental Monitoring [RETS]*

6.0Q11 The TS policy criteria can not be use to relocate programmatic details or requirements from the design feature and administrative controls TS. Provide justification for changes to existing TS 4.10, 4.10.1, 4.10.2, and 4.10.3 Radiological Environmental Monitoring TS consistent with the content of TS found in 10 CFR 50.36. In addition, Table 3.16-1 referenced in existing TS 4.10.1 is not discussed in the existing TS markup justifications. Provide a markup of the Table and the necessary discussions for any changes to the existing TS.

Response: The RETS addressed by this change are contained in the surveillance

section of the CTS (Section 4.0). Consequently, the TS policy criteria are being applied to surveillance requirements associated with CTS requirements that have been similarly relocated (see response to 6.0Q2 and 6.0Q4). The relocation of Table 3.16-1 is addressed in change 26.i above (see response to 6.0Q4) and marked up in Attachment B, Section 5.0. No changes to the RETS surveillance requirements will be made in their relocation.

- i. TS 4.10.1 and Table 4.10-1 - The requirements for the radiological environmental program which provides measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- ii. TS 4.10.2 - The requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iii. TS 4.10.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

38. Tech Specs 4.11 *Refueling Surveillances*

- 6.0Q12 What licensee-controlled document will items of the spent fuel pool charcoal adsorbers tests proposed to be relocated? What mechanism will be used to control future changes to these requirements?

Provide a justification for adding "penetration and system bypass" to the charcoal absorber in-place freon test in proposed specification 5.5.10.d.2. Provide a justification for using "absorber" interchangeably with "adsorber" in proposed TS 5.5.10.d. Explain the use of both "adsorber" and "absorber" in existing TS 4.11.1.1.a.

Response: The deleted text in CTS 4.11.1.1 and CTS 4.11.1.2 marked with a "38.i" in the left margin only reiterate the requirements contained in RG 1.52, Revision 2 which is specifically referenced the VFTP. The deleted testing requirements of "when tested at least 150°F..." will be relocated to station procedures under control of the Ginna procedure change process. All differences between the proposed VFTP and the NUREG-1431 version are being addressed by Plant Systems in their review of 24 month cycles (see RG&E letter dated October 18, 1995). All uses of "absorber" should be replaced with "adsorber" as suggested by the ITS Writer's Guide. The interchangeable use of this in the CTS is a typographical inconsistency. Comment #152 has been opened to address this in the ITS.

- i. TS 4.11.1 - The requirements denoting the Frequency and conditions of the SFP filtration system tests were not added. The level of detail is relocated to the VFTP described in new Specification 5.5.10. [R] This is a Ginna TS Category (iii) change.
- ii. TS 4.11.1.1.a, 4.11.1.1.b, and 4.11.1.1.c - These charcoal adsorber system testing requirements were relocated to the VFTP described in the Administrative Controls (TS 5.5.10). This is a Ginna TS Category (i) change.

39. Tech Specs 4.12 Effluent Surveillances [RETS]

6.0Q13 The TS policy criteria can not be use to relocate programmatic details or requirements from the design feature and administrative controls TS. Provide justification to relocate existing TS 4.12.1, "Liquid Effluents", "4.12.2 "Gaseous Wastes", and 4.12.3, "Waste Gas Decay Tanks" requirements consistent with the content of TS found in 10 CFR 50.36. In addition, Table 3.16-1 referenced in existing TS 4.10.1 is not discussed in the existing TS markup justifications. Provide a markup of the Table and the necessary discussions for any changes to the existing TS.

Response: The RETS addressed by this change are contained in the surveillance section of the CTS (Section 4.0). Consequently, the TS policy criteria are being applied to surveillance requirements associated with CTS requirements that have been similarly relocated (see responses to 6.0Q1, 6.0Q2, and 6.0Q4). The relocation of Table 3.16-1 is addressed in change 26.i above (see response to 6.0Q4) and marked up in Attachment B, Section 5.0. No changes to the RETS surveillance requirements will be made in their relocation.

- i. TS 4.12.1.1 and Table 4.12-1 - The requirements for radioactive material released in liquid effluents to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20,

Appendix B, Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- ii. TS 4.12.1.2 - The requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, GDC 60 and 10 CFR Part 50, Appendix I, Section II.D, were not added. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iii. TS 4.12.2.1 and Table 4.12-2 - The requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iv. TS 4.12.2.2 - The requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- v. TS 4.12.3 - The requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is

not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

40. Tech Spec 4.13 - *Radioactive Material Source Leakage Tests*

6.0Q14 The TS policy criteria can not be use to relocate programmatic details or requirements from the design feature and administrative controls TS. Provide justification to relocate existing TS 4.13, "Radioactive Material Source Leakage Tests requirements consistent with the content of TS found in 10 CFR 50.36. Provide a markup of the Table and the necessary discussions for any changes to the existing TS.

Response: The requirements addressed by this change are contained in the surveillance section of the CTS (Section 4.0). The TS policy criteria are being applied to surveillance requirements which are not associated with any design feature or administrative controls in the CTS or required to be in these sections by 10 CFR 50.36 (see also response to 6.0Q22). Therefore, the TS policy statement can be used as proposed.

- i. TS 4.13 - The requirements for periodic testing of leakage for radioactive sources were not added. The source leak test are not assumed in the analysis of any DBA or transient. Further, the leakage from radioactive sources is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

41. Tech Spec 4.14 - *Snubber Surveillance Requirements*

6.0Q15 Show which programmatic controls and procedural details of TS 4.14 are included in the proposed inservice inspection and inservice test program TS and identify current TS requirements that will be controlled outside TS. [R,T]

Response: See response to 6.0Q3.

- i. TS 4.14 - The requirements for the testing of snubbers were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to Inservice Testing Program described in new Specification 5.5.8 and more generic program description is provided. This is a Ginna TS Category (iii) change.

Administrative Controls

49. Tech Spec 6.1 - *Responsibilities*

- i. TS 6.1.1 - The requirement was revised to include a statement that the Plant Manager shall approve each proposed test, experiment or modification to structures, systems or components that affect nuclear safety. This is a Ginna TS Category (iv.a) change. [M]
- ii. TS 6.1 - A new requirement (Specification 5.1.2) was added which establishes the requirement for Shift Supervisor responsibility. This is a Ginna TS Category (iv.a) change.

50. Technical Specification 6.2 - *Organization*

- i. Cross references to existing regulatory requirements are redundant and generally not incorporated into NUREG-1431. This is a Ginna TS Category.(ii) change. [L]
- ii. Plant specific management position titles in the current Technical Specifications are replaced with generic titles.[L] Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed TS 5.3, and compliance details relating to the plant specific management position titles are identified in licensee controlled documents. The two major specific replacements are the generic "Plant Manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the corporate executive responsible for overall plant nuclear safety" in place of the Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (e.g., as the UFSAR or QA Program). This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change. [Get Human Factors Assessment Branch approval]
- iii. TS 6.2.1.d - The requirement describing the capability of training, health physics and quality assurance to have direct access to responsible corporate management to support mitigation of their concerns was not added. Proposed TS 5.2.1.a requires that "lines of authority, responsibility and communication shall be established and defined throughout the highest management levels." The organizational structure is specified in the Ginna Station QA Program. Since changes to the QA Program are controlled by 10 CFR 50.54(a)(3), equivalent control is provided. This is a Ginna TS Category (ii) change. [R]
- iv. TS 6.2.2.b - The requirements describing the required operating crew compositions were not added. These requirements are specified in 10 CFR 50.54(k), (l), and (m) and proposed TS 5.2.2.a, 5.2.2.b, and 5.2.2.e. This is a Ginna TS Category (ii) change. [R]
- v. TS 6.2.2.d - The requirement was revised to clarify that the individual qualified in radiation protection procedures is allowed

to be absent for not more than two hours. This is consistent with the requirements for shift crew composition. This is a Ginna TS Category (v.c) change. [M]

- vi. TS 6.2.2.e - The requirement describing the overtime requirement for plant staff who perform safety related functions was revised to reference a NRC approved program for controlling overtime. This is a Ginna TS Category (vi) change.

6.0Q16 Provide a discussion showing that the proposed changes result in the same limits as the current requirements, or represent an enhanced presentation of the existing TS intent.

Response: CTS 6.2.2.e states that "shift coverage shall be maintained without routine heavy use of overtime." The CTS then specify that this overtime restriction program should apply to personnel performing safety related functions with examples provided. ITS 5.2.2.e states that "the amount of overtime worked by plant staff members performing safety related functions shall be limited and controlled in accordance with a NRC approved program." The only real difference between the CTS and ITS is that: (1) examples of positions affected by the overtime program are not provided in the ITS, and (2) the ITS requires NRC approval of the program being implemented. The first difference is minor and prevents the need for TS changes when job titles change consistent with other changes to the CTS. The second difference is an enhancement since the SER for the current overtime program specifically states that NRC approval of changes is required (attached). The ITS clarify this requirement.

51. Technical Specification 6.3 - Station Staff Qualifications

- i TS 6.3.1 - The reference to the RG&E letter dated December 30, 1980, was replaced with wording considered more appropriate. The current STA program at Ginna Station is discussed in References 40 and 42 and was reviewed and approved by the NRC. The revised wording eliminates the need to revise the Technical Specifications if the STA program is later revised, but still requires NRC approval of these changes. This is a Ginna TS Category (vi) change.

6.0Q17 Provide discussion showing that the proposed changes result in the same limits as the current requirements, or represent an enhanced presentation of the existing TS intent.

Response: CTS 6.3.1 states the STA program must be in accordance with a 1980 RG&E letter. ITS 5.2.2.e specifies the functions provided by the STA, what MODES the STA is to be assigned to the shift crew, and that the STA must meet the qualifications specified within a NRC approved STA program. The referenced 1980 letter in CTS (attached) does not contain these details, only the training requirements of the STA. In addition, this referenced letter is from RG&E, but should actually be a NRC SER or other correspondence which specifically approves the STA program. The NRC approval of the STA program is contained in a January 12, 1982 letter (attached). RG&E

provided additional information to the NRC in three letters dated February 1, 1982, May 14, 1986, and October 12, 1989. The RG&E program does differ from the NRC Policy Statement of Engineering Expertise on Shift as documented in these three letters; hence the reference to a "NRC approved STA program" in the ITS.

52. Technical Specification 6.4 - Training

- i. TS 6.4 - The requirements for a Training Program were not added. The requirements are either adequately addressed by other Section 5.0 administrative controls or are addressed by 10 CFR 55 requirements. This is a Ginna TS Category (ii) change. [R]

53. Technical Specification 6.5

None.

54. Technical Specification 6.6

None.

55. Tech Spec 6.7 Safety Limit Violation

6.0Q18 For each of the Category (ii) changes identify the plant document that includes the duplicate TS requirement.

Response: For a safety limit violation to occur, a plant transient or accident must first occur with subsequent failure of multiple safeguards equipment (i.e., a safety limit violation will not occur during normal power operation). Therefore, the notification of management and NRC parties will be performed during the implementation of the emergency response procedures for an "Alert" or higher.

For CTS 6.7.1.b, procedure EPIP 1-5 (attached) currently requires notification of NRC within 1 hour per 10 CFR 50.72 (Step 4.2) and immediate notification of the Senior VP, Customer Relations (Attachment 2, Step 4.a). Members of the offsite review board (i.e., NSARB) are listed in Attachment 5 of this procedure, but are not specifically identified as being required to be notified. However, since the Senior VP is also chairman of the NSARB, this requirement can be considered met. Therefore, CTS 6.7.1.b has been relocated to procedure EPIP 1-5.

CTS 6.7.1.c and 6.7.1.d relate to development of the Safety Limit Violation Report and the time requirements for submittal to the NRC, offsite review board, and Senior Vice President, Customer Relations. The 14 day limit for submittal of this report to the NRC (CTS 6.7.1.d) is contained in procedure A-25.6, Step 3.5. The content of this report (CTS 6.7.1.c) would be treated similar to an LER which is discussed in Step 3.10 of this procedure.

- i. TS 6.7.1.a - The initial operator actions for Safety Limit (SL) violations were revised as follows:

- a. For violation of the Reactor Core or RCS Pressure SL in MODES 1 and 2, the requirement to immediately shutdown the reactor (effectively to be in MODE 3) was revised to allow 1 hour to restore compliance and place the unit in MODE 3. Immediately shutting down the reactor could infer action to immediately trip the reactor. The revision provides the necessary time to shutdown the unit in a more controlled and orderly manner than immediately tripping the reactor which could result in a plant transient. The proposed time continues to minimize the time allowed to operate in MODE 1 or 2 with a SL not met. This is a Ginna TS Category (v.b.44) change. [L]
- b. For violation of the RCS Pressure SL in MODES 3, 4, and 5, an additional action was added which requires restoring compliance with the SL within 5 minutes. Specifying a time limit for operators to restore compliance provides greater guidance to plant staff. This is a Ginna TS Category (v.a) change. [M]
- ii. TS 6.7.1.b - The requirement for notification to management personnel and the offsite review function of a SL violation was not added to the new specifications. Notification requirements are relocated to the TRM. This is a Ginna TS Category (iii) [R] change. The requirement for notification to the NRC of a SL violation was not added to the new specifications since this requirement is denoted in 10 CFR 50.36 and 10 CFR 50.72. This is a Ginna TS Category (ii) change. [R]
- iii. TS 6.7.1.c - The requirement that a Safety Limit Violation Report be prepared was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the onsite review committee to review the Safety Limit Violation Report was not added to the new specifications. The responsibilities of the onsite review committee are relocated to the TRM. This is a Ginna TS Category (iii) change. SL violations are reported to the NRC in accordance with the provisions of 10 CFR 50.73. The details describing the requirements for content of the Safety Limit Violation Report is, therefore, controlled by the provisions of 10 CFR 50.73 and does not need to be specified in TS. This is a Ginna TS Category (ii) change. [R]
- iv. TS 6.7.1.d - The requirement for the submittal of a Safety Limit Violation Report to the NRC was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. [R] The requirement for the submittal of a Safety Limit Violation Report to management personnel and the offsite review function was not added to the new specifications. The distribution of reports submitted in accordance with 10 CFR 50.73 are relocated to the TRM. This is a Ginna TS Category (iii) change. [R]

56. Technical Specification 6.8 *Procedures*

6.0Q19 For each of the Category (ii) changes discuss identify the plant

document that includes the duplicate TS requirement.

Response: The PCP (CTS 6.8.1) is contained in procedure RPA-RW-PCP (attached). No other items are Category (ii) changes.

- i. TS 6.8.1.d - The Offsite Dose Calculation Manual implementation is covered by a more generic item which is specified in Section 5.5. It is not necessary to specifically identify each program under procedures (see Section D, item 56.iv). Since the requirements remain, this is considered to be a change in the method of presentation only. This is a Ginna TS Category (i) change. [A]
- ii. TS 6.8.1.e - The PCP description was not added since this program only implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71 and does not impose any new regulations. The detailed description of the PCP is provided in licensee controlled documents with the requirement for the PCP relocated to the TRM. This is a Ginna TS Category (ii) change. [R]
- iii. TS 6.8.1 - A new specification (TS 5.4.1.b) was added which establishes the requirement for written emergency operating procedures implementing the requirements of NUREG-0737 and NUREG-0737, Supplement 1. This is a Ginna TS Category (iv.a) change. [A]
- iv. TS 6.8.1 - A new specification (TS 5.4.1.e) was added which establishes the requirement for written procedures for programs and manuals denoted in new Specification 5.5. These Programs include:

<u>ITS</u>	<u>Current TS</u>	<u>Program</u>
5.5.1	1.13 & 6.15	Offsite Dose Calculation Manual
5.5.2	4.4.3	Primary Coolant Sources Outside Containment
5.5.3	New	Post Accident Sampling Program
5.5.4	3.9 & 3.16	Radioactive Effluent Controls Program
5.5.5	New	Component Cyclic or Transient Limit
5.5.6	4.4.4	Pre-Stressed Concrete Containment Tendon Surveillance
		Program
5.5.7	New	Reactor Coolant Pump Flywheel Inspection Program
5.5.8	4.2	Inservice Testing Program
5.5.9	4.2	Steam Generator (SG) Tube Surveillance Program
5.5.10	4.5.2.3 & 4.11.1	Ventilation Filter Testing Program
5.5.11	3.9.2.5 & 3.9.2.6	Explosive Gas and Storage Tank Radioactive Monitoring Program
5.5.12	4.6.1.d	Diesel Fuel Oil Testing Program
5.5.13	New	Technical Specification Bases Control
5.5.14	New	Safety Function Determination Program

The technical content of several requirements are being moved from other chapters of the current Technical Specifications and are proposed to be identified as Programs in accordance with the format of NUREG-1431. This is a Ginna TS Category (i) change. Other programs were added, except as discussed below, to ensure

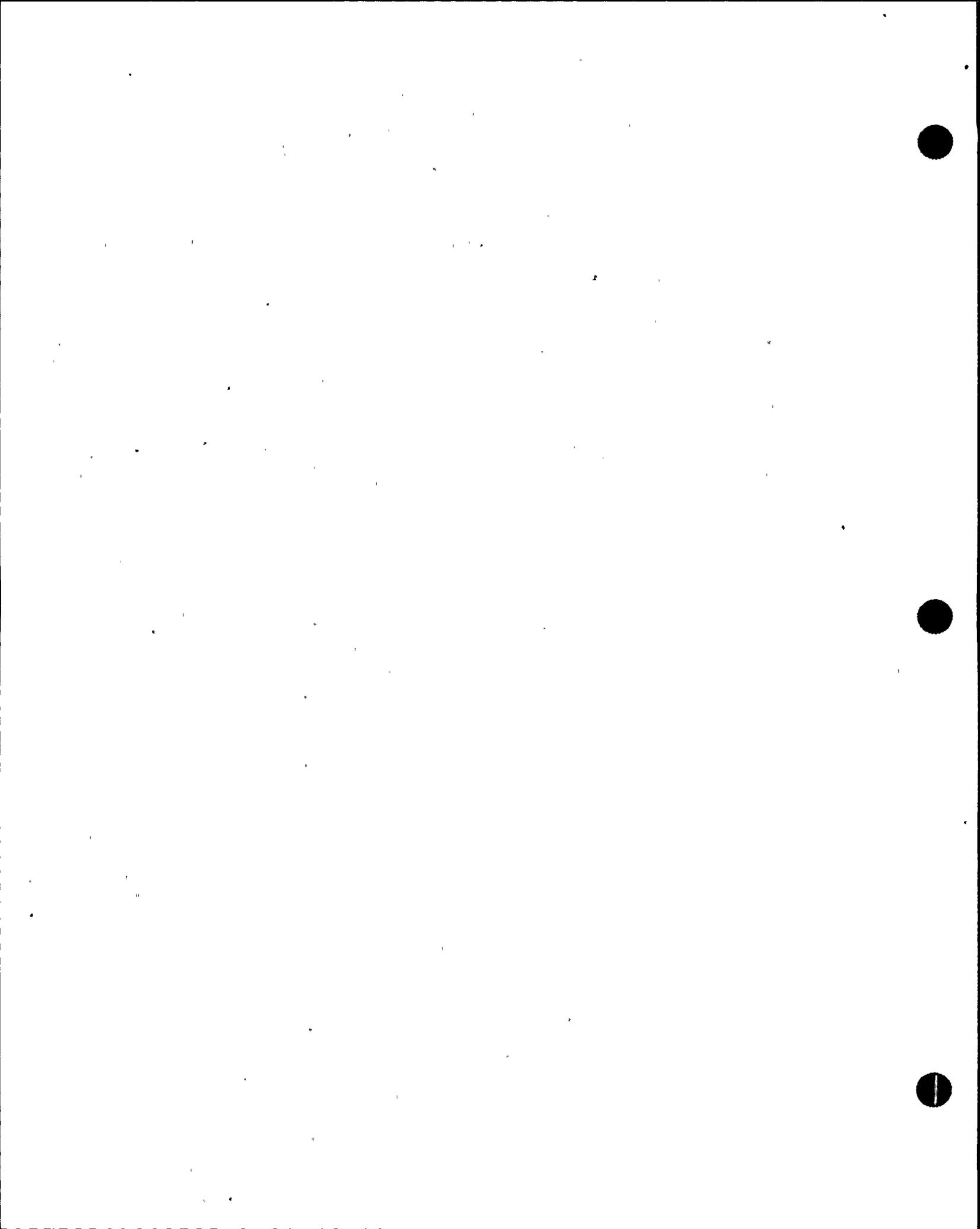
consistency in the implementation of required programs within the current licensing basis. The Radioactive Effluent Controls Program was added due to the relocation of the radiological Technical Specifications consistent with Generic Letter 89-01 and the changes to 10 CFR 20. The Bases Control program was added to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program was added to support implementation of the support system operability characteristics of the Technical Specifications (new LCO 3.0.6). These are Ginna TS Category (iv.a) changes. [A]

57. Technical Specification 6.9 - *Reporting Requirements*

6.0Q20 For each of the Category (ii) changes identify the plant document that includes the duplicate TS requirement.

Response: CTS 6.9.1.1 relates to the requirement for a Startup Report following (1) receipt of an operating license, (2) increased power level, (3) change in nuclear fuel design, and (4) significant plant modifications. Ginna has already received its operating license such that item (1) no longer applies. An increase in power level would require a change to the TS and operating license to implement while changes in nuclear fuel design and significant plant modifications could possibly require a TS change. If a TS change is required for items (2), (3), and (4), then NRC approval must be obtained prior to implementation such that the report provides little benefit. If a TS change is not required for items (3) and (4), then documentation of these changes would be performed in accordance with 10 CFR 50.71 (i.e., UFSAR updates). The implementation of UFSAR updates is performed by procedure EP-2-P-112 (Step 5.3.3.e for NRC copies). No other Category (ii) changes exist in this section.

- i. TS 6.9 - The reference to reporting requirements were revised consistent with 10 CFR 50.4. This is a Ginna TS Category (vi) change. [A]
- ii. TS 6.9.1.1 - The requirement to submit a Startup Report was not added. The Startup Report is more appropriately addressed in the NRC Safety Evaluation Report authorizing an Operating License, increased power level, installation of a new nuclear fuel design or manufacturer, or modifications which significantly alter the nuclear, thermal, or hydraulic performances of the plant. The Startup Report is required to be submitted within 90 days following completion of the above activities and does not require NRC approval. Therefore, inclusion of the requirement for this report in Technical Specifications is not necessary to assure safe plant operation. This is a Ginna TS Category (ii) change. [R]
- iii. TS 6.9.1.2 - The requirements describing the details of the monthly report were not added. These details are appropriately relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change. [R]



6.0Q21 Provide discussion for the changes shown as a markup of existing TS 6.9.1.2.

Response: The changes to the third, fourth, and fifth lines of CTS 6.9.1.2 are "pen and ink" changes that were performed based on verbal discussions with the Ginna project manager following changes to 10 CFR 50.4 (i.e., all official copies of the CTS contain these written changes). Therefore, these changes are not discussed in the conversion to ITS.

- iv. TS 6.9.1.3, TS 6.9.1.4, Table 6.9-1 and Table 6.9-2 - The details and methods implementing these specifications were not added. These details are appropriately relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. The submittal date was also changed to May 15th to allow the submittal of the Annual Radiological Environmental Operating Report to correspond with the Monthly Operating Report submittal date. This is a Ginna TS Category (iii) change. [R]
- v. TS 6.9.1.4 - The specific date referenced for the annual submittal was revised consistent with the requirements of 10 CFR 50.36a. This is a Ginna TS Category (vi) change. [A]
- vi. TS 6.9.1.5 - The requirement for the reporting of challenges to pressurizer PORVs or safety valves was revised from an annual to a monthly report and relocated to the Monthly Operating Report (new Specification 5.6.4). This is a Ginna TS Category (v.c) change. [A]
- vii. TS 6.9.2.1 - The reporting requirement related to sealed sources was not added since this is specified in 10 CFR 30.50. This is a Ginna TS Category (ii) change. [R]

6.0Q22 For the Category (ii) change identify the plant document that includes the duplicate TS requirement.

Response: CTS 6.9.2.1 specifies reporting requirements if leak testing of sealed sources indicates leakage ≥ 0.005 microcuries or removable contamination. Procedure HP-8.2 contains the leak testing requirements for sealed sources. The requirement for NRC notification is contained in procedure CHA-RETS-REP-ANNUAL (step 9.1 and Attachment I, 11) (attached).

- viii. TS 6.9.2.4 - The reporting requirement for reactor overpressure protection system operation was revised. The reporting requirement is detailed in proposed Specification 5.6.4, and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. Since the criteria identified in 10 CFR 50.73 includes the area of degraded boundaries that necessitates reporting, any minor differences are negligible with regard to safety. This is a Ginna TS Category (v.c) change. [R]

6.0Q23 Provide discussion showing that the proposed changes result in the

same limits as the current requirements, or represent an enhanced presentation of the existing TS intent.

Response: CTS 6.9.2.4 requires a special report to be submitted in 30 days in the event the LTOP system is actuated to mitigate a RCS pressure transient. The CTS also requires that the report document: (1) the circumstances initiating the event, (2) the effect of the LTOP system, and (3) any corrective actions taken. ITS 5.6.4 requires a monthly report by the 15th of each month that includes documentation of all challenges to the PORVs or pressurizer safety valves. Therefore, the report submittal time requirements remain the same. The monthly operating report required by ITS 5.6.4 does not have the same level of detail as CTS 6.9.2.4; however, this report would document items (1) and (2) above based on CTS 6.9.1.2 and NRC guidance. In addition, 10 CFR 50.73 addresses all three CTS documentation requirements within 30 days of the event. Therefore, there is equivalent requirements in the CTS and the ITS and existing regulations.

- ix. A new requirement TS 5.6.5 was added which establishes the reporting requirement for the COLR. The COLR is required due to the removal of existing Technical Specification core operating limits. This is a Ginna TS Category (iv.a) change. [M]
- x. A new requirement TS 5.6.6 was added which establishes the reporting requirement for the RCS PTLR. The PTLR is required due to the removal of existing Technical Specification pressure and temperature operating limits. This is a Ginna TS Category (iv.a) change. [M]

58. Technical Specification 6.10

None.

59. Technical Specification 6.11

None.

60. Technical Specification 6.12

None.

61. Technical Specification 6.13

- i. TS 6.13.1 - Plant specific position titles in the current Ginna Station TS were replaced with generic titles. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (e.g., the UFSAR or QA Program). This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change. [L]

6.0Q24 Provide discussion showing that the proposed changes result in the same limits as the current requirements, or represent an enhanced presentation of the existing TS intent.

Response: CTS 6.13.1 contains a note which states that alternate titles may be specified for a position as long as all TS requirements continue to apply to the alternate title and that the alternate titles are specified in the UFSAR. Consequently, the CTS essentially use generic titles now since even though a specific title is provided, a note allows this title to change. The requirement to specify titles in the UFSAR is addressed by ITS 5.2.1.a.

62. Technical Specification 6.14

None.

63. Technical Specification 6.15

- i. TS 6.15.1.b - The approval process for ODCM changes was revised to clarify that the effective changes be approved by the Plant Manager instead of the onsite review function. Since the onsite review function reports to the Plant Manager, this is a conservative change. This is a Ginna TS Category (v.c) change. [A]

64. Technical Specification 6.16

- i. TS 6.16 - The process for changes to the PCP was not added to the new specifications since this program only implements the requirements of 10 CFR Part 20, 10 CFR Part 61, and 10 CFR Part 71 and does not impose any new requirements. The detailed description of the PCP is provided in licensee controlled documents and the requirement for the program is relocated to the TRM. This is a Ginna TS Category (ii) change. [R]

6.0Q25 For the Category (ii) change discuss identify the plant document that include the duplicate requirement.

Response: See response to 6.0Q19.

65. Technical Specification 6.17

- i. TS 6.17 - The requirements for major changes to radioactive waste treatment systems was not added. Changes to these systems are controlled by 10 CFR 50.59. NRC notification of significant changes to these systems is addressed by 10 CFR 50.59(b)(2). Therefore, this specification is relocated to the TRM. This is a Ginna TS Category (iii) change. [R]

6.0Q26 For the Category (iii) change identify the plant document that includes the duplicate requirement.

Response: The review of all plant changes under 10 CFR 50.59 is contained in procedures IP-SEV-1 and IP-SEV-2 which have been previously provided with respect to procedure controls. In addition, procedure CHA-

RETS-REP-ANNUAL (step 9.1 and Attachment I, 14.0) contains the requirement to document major changes to the radioactive waste treatment system in the annual Radioactive Effluent Release Report (ITS 5.6.3).

Section 6.0 Improved TS

114. ITS 5.1

- i. Incorporation of approved Traveller BWO-09, C.1.
- ii. Incorporation of approved Traveller NRC-02, C.21.
- iii. Incorporation of approved Traveller BWO-09, C.2.

115. ITS 5.2

- i. TS 5.2.1.c - This section describing the capability of training, health physics and quality assurance to have direct access to responsible corporate management to support mitigation of their concerns was not added. TS 5.2.1.a requires that "lines of authority, responsibility and communication shall be established and defined throughout the highest management levels." The organizational structure is specified in the Ginna Station QA Program. Since changes to the QA Program are controlled by 10 CFR 50.54(a)(3), equivalent control is provided. This is an ITS Category (iii) change.

6.0Q27 Provide a submittal to correct the referenced TS to TS 5.2.1.d. This proposed change is a generic relaxation that requires an approved NEI traveler. Provide the RGE policy that implements this administrative control.

Response: RG&E agrees to withdraw this change. Comment #153 has been opened to add this specification back into the ITS.

- ii. Incorporation of approved Traveller BWO-09, C.3.
- iii. TS 5.2.2.b - This section describing the required operating crew compositions was not added. These requirements are specified in 10 CFR 50.54(k), (l), and (m) and proposed TS 5.2.2.a, 5.2.2.b, and 5.2.2.e. This is an ITS Category (iii) change.

6.0Q28 Provide discussion explaining in more detail the equivalence of the referenced proposed TS and the regulations to the NUREG.

Response: NUREG Specification 5.2.2.b states that:

At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

The first sentence of 10 CFR 50.54(m)(2)(iii) states that "when a nuclear power unit is in an operational mode other than cold

shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times." This is equivalent to the last sentence of Specification 5.2.2.b. The second sentence of 10 CFR 50.54(m)(2)(iii) states that "in addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." This is equivalent to the first sentence of Specification 5.2.2.b.

- iv. Incorporation of approved Traveller BWOG-09, C.4.
- v. TS 5.2.2.e - This section describing the overtime requirement for unit staff who perform safety related functions to require control in accordance with an NRC approved program was revised. RG&E currently utilizes a staff working hour control program which slightly differs from the NRC Policy Statement on Working Hours (Generic Letter 88-12). This program was previously reviewed and approved by the NRC (Ref. 40). The proposed wording is considered more appropriate and consistent with the current Technical Specifications. This is an ITS Category (i) change. [A]
- vi. TS 5.2.2.f - This section describing the requirements for the Operations Manager to hold an SRO license was not added. The qualifications of this position are addressed in ANSI Standard N18.1-1971 referenced in TS 5.3. This is an ITS Category (i) change.
- vii. TS 5.2.2.g - This section describing the requirements of the Shift Technical Advisors (STAs) was revised. The requirements specified in TS 5.3.1 are moved to TS 5.2.2.g in accordance with approved Traveller BWOG-09, C.6. The wording of Traveller BWOG-09, C.6 was revised to reflect more appropriate and consistent wording to Ginna Station commitments. The STA program does not meet all the requirements denoted in the Commission Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04). The current STA program is discussed in References 41 and 42 and was reviewed and approved by the NRC. This is an ITS Category (i) change.

6.0Q29 Revise proposed TS 5.2.2.g to incorporate the specific information in Refs. 41 and 42 in place of the Commission Policy Statement on Engineering Expertise on Shift.

Response: Ginna has several differences from the Commission Policy Statement including:

- a. RG&E continues to use a dedicated STA separate from the shift supervisor as proposed in the policy statement.
- b. The RG&E educational requirements for the STA are less stringent than required by the policy statement.

These differences have been reviewed by the NRC and found to "meet the intent" of the STA requirements (see attached letters). This

level of detail is not in the current TS and RG&E does not believe its necessary to specify this in the Administrative Controls. Instead, requiring the program to be one that is NRC approved allows future changes to the program without requiring unnecessary TS changes.

116. ITS 5.3

- i. TS 5.3.1 - The requirement for qualifications of staff not covered by Regulatory Guide 1.8 was not added. This requirement was not considered necessary since all activities which affect nuclear safety are controlled by other technical specification requirements, existing regulations, and the QA Program. Also, Revision 1 of Regulatory Guide 1.8 was not revised to Revision 2 in order to maintain consistency with the current QA Program and existing procedures. This is an ITS Category (i) change.

6.0Q30 For the category (i) change explain all changes to the current TS requirements showing how the proposed TS limits are the same as the current limits.

Response: CTS 6.3.1 states that "each member of the facility shall meet or exceed the minimum qualifications of ANSI Standard N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," as supplemented by Regulatory Guide 1.8, September 1975, for comparable positions, except for the Shift Technical Advisor." ITS 5.3.1 states that "each member of the plant staff shall meet or exceed the minimum qualifications of ANSI Standard N18.1-1971, as supplemented by Regulatory Guide 1.8, September 1975, for comparable positions." As can be seen, the only difference between the two requirements is the discussion of the STA in the CTS. This singular difference is discussed in the response to 6.0Q17.

- ii. Incorporation of approved Traveller BWOG-09, C.6.

117. ITS 5.4

- i. Incorporation of approved Traveller BWOG-09, C.7.

118. ITS 5.5

- i. Incorporation of approved Traveller BWOG-09, C.8.

119. ITS 5.6

- i. Incorporation of approved Traveller BWOG-09, C.9.
- ii. Incorporation of approved Traveller WOG-06, C.1, was modified due to the format changes provided by Traveller BWOG-09, C.9.

120. ITS 5.7 - *Program and Manuals*

- i. Incorporation of approved Traveller BWOG-09, C.10.

- ii. Incorporation of approved Traveller WOG-06, C.7.
- iii. Incorporation of approved Traveller BWOG-09, C.11, supersedes changes proposed by approved Travellers WOG-06, C.2, and WOG-06, C.3.
- iv. Incorporation of approved Traveller BWOG-09, C.12.
- v. Incorporation of approved Traveller BWOG-09, C.13.
- vi. Incorporation of approved Traveller BWOG-09, C.13, supersedes changes proposed by approved Traveller WOG-06, C.3. Additional cross references, similar to those deleted by Traveller BWOG-09, C.13, were not added. In general, the format of the NUREG-1431 does not include the use of cross references. This is an ITS Category (iv) change.
- vii. These changes are provided for consistency with the new 10 CFR 20 references. This is an ITS Category (iv) change.

6.0Q31 Confirm that the NRC staff accepts the new 10 CFR Part 20 Ginna licensing basis.

Response: The new 10 CFR Part 20 was required to be implemented by all licensees by January 1, 1994. The NRC does not have to approve this program for each licensee. Instead, the NRC inspects and audits compliance with the new 10 CFR Part 20. Recent inspections for Ginna include 94-08 and 95-05 (attached).

- viii. Incorporation of approved Traveller BWOG-09, C.14.
- ix. Incorporation of approved Traveller BWOG-09, C.15.
- x. Incorporation of approved Traveller BWOG-09, C.16.
- xi. Incorporation of [approved] Traveller BWOG-09, C.17.
- xii. TS 5.7.2.13 - The requirements for the Steam Generator (SG) Tube Surveillance Program were revised to reflect current Ginna Station licensing basis. Incorporation of approved Traveller BWOG-09, C.18, provided a reviewers note that the licensees current licensing basis program description be provided. The proposed TS 5.5.9 provides this program description. This is an ITS Category (iv) change.

6.0Q32 Provide a source document reference for the SG Tube Surveillance Program description used in proposed TS 5.5.9.

Response: The source document for ITS 5.5.9 is CTS 4.2.1.4.

- xiii. Incorporation of approved Traveller WOG-06, C.4.
- xiv. TS 5.7.2.15 - The requirements for the Ventilation Filter Testing Program (VFTP) were revised to reflect current Ginna Station test frequencies and methods. These are performed, where practical, in

accordance with Regulatory Guide 1.52 and ANSI N510-1975. Due to the revision of TS 5.7.2.15, the approved traveller WOG-06, C.5, was not incorporated. This is an ITS Category (i) change.

6.0Q33 Provide documentation that proposed TS 5.5.10 changes result in the same limits as the current TS limits. Also, include documentation of the proposed changes to the NUREG that delete the provisions of SR 3.0.2 and SR 3.0.3

Response: ITS 5.5.10 is the same as CTS 4.5.2.3 except as discussed in the responses to 3.6Q9 and 3.6Q10. The deletion of the SR 3.0.2 and SR 3.0.3 statements in Attachment D is a typographical error since this statement is included in Attachment C. Comment #154 has been opened to correct this error in Attachment D.

xv. Incorporation of approved Traveller NRC-02, C.13.

xvi. TS 5.7.2.16 - The requirement for control of the quantity of radioactivity contained in outdoor liquid radwaste tanks was not added since there are no outdoor liquid radwaste tanks at Ginna Station. The description of the methodology used in determining radioactivity quantities in the waste gas decay tanks was revised to reflect current licensing basis. This is an ITS Category (i) change.

6.0Q34 Provide documentation that the proposed TS 5.5.11, Explosive Gas and Storage Tank Radioactivity monitoring program results in the same limits as current TS.

Response: The differences between ITS 5.5.11 and CTS 3.9.2.5 are discussed in change 19.viii. However, to expand this discussion, the following is also provided. The requirement to limit the oxygen concentration limits in the waste gas decay tanks is relocated to ITS 5.5.11 but the actual limits are to be controlled by procedure CH-SAMP-MSA outside TS (attached). The requirement for a surveillance program to verify these limits are met is a new TS requirement for Ginna. However, this surveillance program is also addressed by procedure CH-SAMP-MSA.

xvii. TS 5.7.2.12 - The inservice testing program description was revised to include high energy piping outside containment and steam generator tubes. This is consistent with the Ginna Station current licensing basis and approved IST program. This is an ITS Category (ii) change.

6.0Q35 Provide documentation that the proposed TS 5.5.4.b, limitations on liquid effluent releases to unrestricted areas and proposed TS 5.5.8, ITS Program results in the same limits as current TS.

Response: The differences between ITS 5.5.4.b (RETS) and CTS is addressed in the response to 6.0Q1; 6.0Q2, 6.0Q4, 6.0Q5, and 6.0Q11. The differences between ITS 5.5.8 and CTS is addressed in the response to 6.0Q6.

xviii. TS 5.7.2.17 - The requirement denoting the purpose of the diesel

fuel oil testing program was revised to reflect Ginna Station current licensing basis. Approved Traveller WOG-06, C6, was not incorporated due to the proposed revisions to these requirements. This is an ITS Category (i) change.

6.0Q36 Provide documentation that the proposed TS 5.5.12, diesel fuel oil testing program results in the same limits as current TS.

Response: CTS 4.6.1.d requires verification every 92 days that "a sample of diesel fuel from the fuel storage tanks is within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment." ITS 5.5.12 requires a program with sampling, testing, and acceptance criteria of Table 1 of ASTM D975. The differences between the CTS and ITS include: (1) the relocation of the frequency of this fuel oil sample to procedure CH-S-F0 outside TS (attached), and (2) the year of the ASTM standard is not provided in the ITS. The year was omitted to allow a change to a more recent ASTM standard without requiring a TS change. It should be noted that the CTS only require checks for viscosity, water and sediment while the ITS requirement is more "open ended." However, Table 1 of ASTM D975 includes checks of flash points, ash, sulfur, copper, etc. which RG&E is not agreeing to perform. Therefore, RG&E proposes to revise ITS 5.5.12 to only require checks of viscosity, water and sediment. This is consistent with the CTS and the bases for ITS SR 3.8.3.2. Comment #155 has been opened to address this. [This response was changed as a result of meetings the week of 11/1/95. See comment #196.]

xix. Incorporation of approved Traveller BWOG-09, C.19.

xx. TS 5.7.2.14 - The secondary water chemistry program was revised consistent with the current program specified in the Ginna Station license. These are ITS Category (i) changes.

6.0Q37 Provide the correct reference in place of reference 120.xx on page 5.0-21, - 5.0-25 in the NUREG markup as necessary. Provide documentation that proposed TS 5.5.2, and proposed TS 5.5.5 [need the FSAR reference] result in the same limits as the current TS. Include an explanation of why the proposed deletion of the ALARA program objective does not apply to Ginna. Provide a source document reference for the Secondary Water Chemistry Program description used in proposed TS 5.5.15.

Response: Change "120.xxi" should be provided in place of change "120.xx" on the following pages: 5.0-20, 5.0-21, 5.0-23, and 5.0-25 (attached). This new change justification is provided below:

120.xxi Various editorial changes were made within the Administrative Controls Programs and Manual section. These include adding "program" after the title to Primary Coolant Sources Outside Containment, Post Accident Sampling, and Component Cyclic or Transient Limit to be consistent with the rest of the section. Also, the Component Cyclic or Transient Limit Program was revised to only generically reference the UFSAR without identifying a specific section

to prevent a TS change if the UFSAR section is renumbered. (Note - the UFSAR is currently undergoing a major rewrite as result of the new SGs, ITS, and 18 month cycles.) Finally, the use of "TS" in Specification 5.5.8.d was replaced with "Technical Specification" since this abbreviation is not previously defined in this section. These are ITS Category (iii) changes.

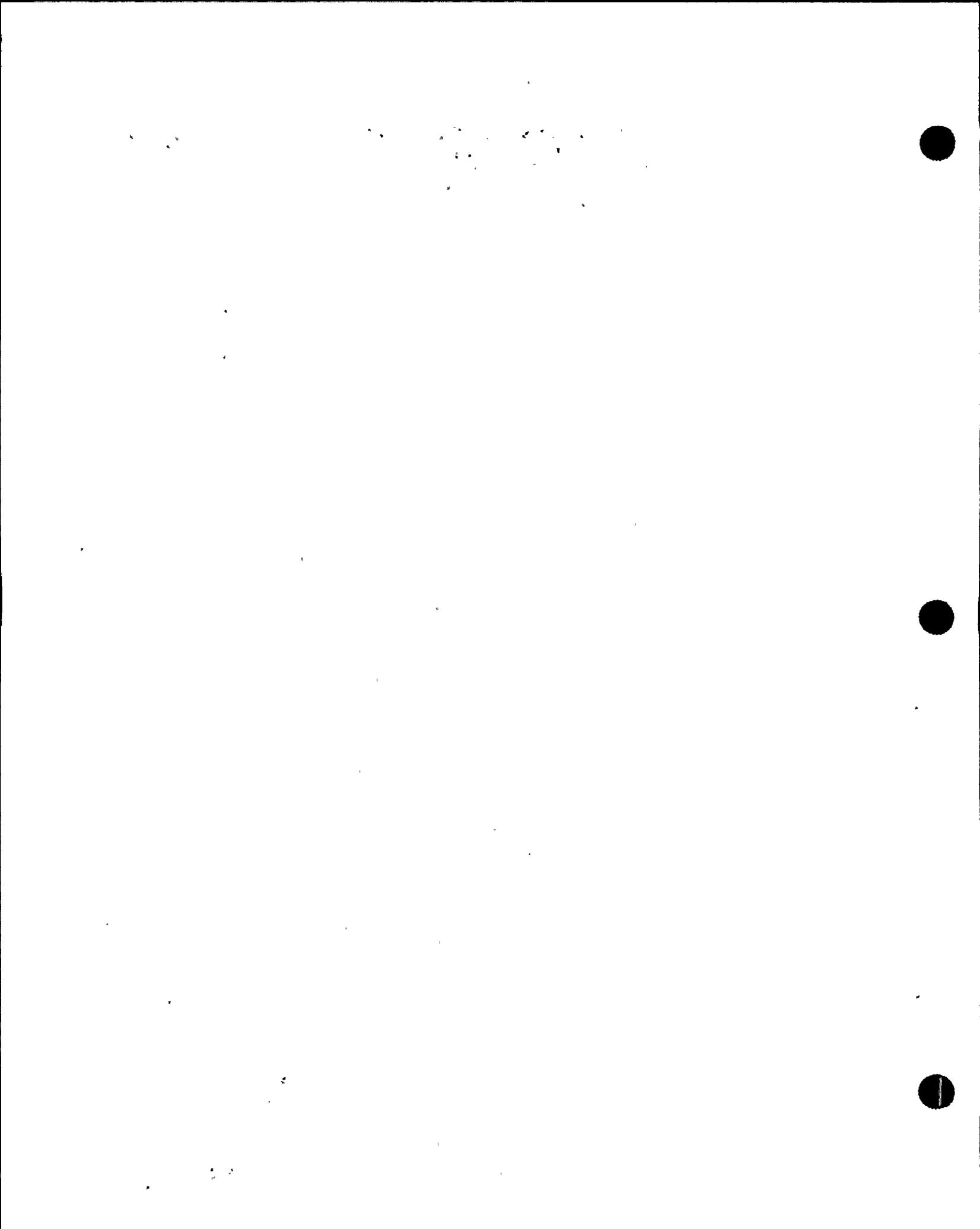
Also, the following new change justification should be added for page 5.0-20.

120.xxii The ALARA requirements of the Primary Coolant Sources Outside Containment program were not added to the new specifications. The dose analysis for Ginna uses a total leakage from all affected systems of 2 gallons per hour (CTS 4.4.3.2) since no credit is taken for the Auxiliary Building Ventilation System following an LOCA. This value is proposed to be controlled by station procedures following relocation from the CTS. Therefore, specifying ALARA is unnecessary and in some instances, could exceed the accident analysis assumptions. This is an ITS Category (i) change.

With respect to the comparison of ITS 5.5.2 and CTS 4.4.3, see the response to 6.0Q8. The source document for the Secondary Water Chemistry Program is the Ginna license, page 4.

Finally, as a result of the previous "missing" change justifications identified above, a comparison of the NUREG Markup (Attachment D) and the retyped version (Attachment C) was performed. This review identified two items on the retype that were not in the NUREG markup. These changes to the NUREG markup are provided using yellow highlights to show the changes. The first item relates to Insert 5.0.5 and does not require additional justification. The second item relates the SFDP which does require a justification as provided below [This response was changed as a result of meetings the week of 11/1/95. See comment #187.]:

121.iii Editorial changes were made to the SFDP to provide additional clarity. These changes were requested by operational personnel to better understand the intent of this program. The most significant change is to clarify that a loss of safety function can exist at a train level and not just due to system inoperability as the NUREG states. For example, CCH Pump A (as supplied by diesel generator A) could be inoperable in accordance with LCO 3.7.7 but the CCH system itself remains OPERABLE (i.e., you're in a required action with 72 hours to complete). If diesel generator B were subsequently declared inoperable, a loss of safety function exists upon a loss of offsite power. This loss of function is due to the loss of two trains of two systems. This is an ITS Category (iii) change.



Also, additional changes that should have been made to the both the markup and the retype were identified on pages 5.0-23 and Insert 5.0.5 (attached and highlighted). No further justifications are required for these changes but Comment #156 has been opened to track their incorporation. [End]

121. ITS 5.8

- i. Incorporation of approved Traveller BWOG-09, C.20.
- ii. Incorporation of approved Traveller BWR-25, C.3.

122. ITS 5.9

- i. Incorporation of approved Traveller BWOG-09, C.21.
- ii. The incorporation of approved Traveller BWOG-09, C.21, was revised to reflect a submittal date consistent with the reporting requirements of 10 CFR 20.2206(c). This is an ITS Category (iv) change.

6.0Q38 Provide documentation that the proposed TS 5.6.1 results in the same limits as current TS.

Response: The only difference between CTS 6.9.2.2 and ITS 5.6.1 is that: (1) the ITS reference the new part 20 section (20.2206(c) versus 20.407), and (2) the parenthetical "(describe maintenance)" was removed in the ITS. This parenthetical is not necessary to describe the type of job functions required to be addressed in the annual report.

- iii. Incorporation of approved Traveller BWOG-09, C.22.
- iv. Incorporation of approved Traveller BWOG-09, C.23.
- v. Incorporation of approved Traveller BWR-06, C.7.
- vi. TS 5.9.2.b - The requirement for a special report following four or more valid failures of an individual emergency diesel generator in the last 25 demands was not added since the requirement is not specified in the current Technical Specifications. Any required report can be adequately controlled by the licensees administrative controls. This is an ITS Category (i) change.

6.0Q39 Provide discussion explaining a technical or hardship basis for not choosing to adopt the NUREG TS.

Response: RG&E does not have any requirements for accelerated testing of the diesel generators and is not willing to add this requirement nor special report. This is described in Attachment A, Section C, item 94.viii. Accelerated testing has been demonstrated to cause additional diesel generator wear and tear and subsequent reduction in system reliability. In addition, the NRC has allowed this testing and special report to be eliminated upon implementation of

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the Maintenance Rule. RG&E plans to implement the Maintenance Rule by June 1, 1996 which is only 5 months after implementation of the ITS. This small time difference is considered minor.

- vii. TS 5.9.2.d - The requirement for a special report following degradation of the containment structure detected during test required by the Pre-stressed Concrete Containment Tendon Surveillance Program was not added since the requirement is not specified in the current Technical Specifications. Any required report can be adequately controlled by the licensees administrative controls. This is an ITS Category (i) change.

6.0Q40 Provide discussion explaining a technical or hardship basis for not choosing to adopt the NUREG TS.

Response: This special report is only required by the ITS in the event of "any abnormal degradation of the containment structure." However, "abnormal degradation" is not defined leaving this open to interpretation by the NRC and RG&E. Also, there is no time limit on submitting this special report. Also, if RG&E commits to RG 1.35 (see response to 6.0Q7), section 8 of this document reiterates this special report requirement. This special report is also not contained in the CTS. 10 CFR 50.73(v) requires reporting of "any event of condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (e) control the release of radioactive material." Therefore, if the tendon surveillance program identified sufficient degradation to affect containment OPERABILITY, a LER is required. RG&E considers this sufficient reporting requirements without requiring a special report.

- viii. Incorporation of approved Traveller BWOG-09, C.18. This Traveller was revised to reflect that the requirement for a special report for steam generator tube inspections was not added since the requirement is not specified in the current Technical Specifications. Any required report can be adequately controlled by the licensees administrative controls. This is an ITS Category (i) change.

6.0Q41 Provide discussion explaining a technical or hardship basis for not choosing to adopt the NUREG TS.

Response: The requirement for the special report is in the ISI Program (attached) and not in the CTS. Requiring this to be specified in the ITS is only redundant and unnecessary. The ISI Program requires NRC review and approval for changes.

- ix. TS 5.9.2.c - The requirement for a Special Report following extended Post Accident Monitoring instrumentation inoperability and the associated details of the report and when it should be submitted were not added. The details can be adequately controlled by the licensee's administrative controls. This information was added to the bases for the LCO Required Actions which required the Special Report to be written. This is an ITS Category (iii) change.

6.0Q42 This report is necessary to the proper application of the PAM

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instrumentation TS. Provide a redraft of the PAM TS and the associated Bases that moves the report to an LCO Action condition for failure to meet channel operability requirements for radiation monitors located inside containment. Otherwise the alternative is to initiate a plant shutdown.

Response: The PAMS ITS includes this special report (see Required Actions C.1 and F.1 of ITS LCO 3.3.3). Suggest this issue be addressed during review of this LCO.

6.0Q43 Comment 122.x is used in the NUREG markup. Provide the discussion.

Response: Change justification 122.x is provided below:

122.x The reference to the Process Control Program was not added to the Radioactive Effluent Release Report. This program is no longer defined in the NUREG as it was removed via BWO-09, C.13. Maintaining reference to this program creates the potential for confusion without any additional benefit. This is an ITS Category (iii) change.

123. ITS 5.10

- i. Incorporation of approved Traveller BWO-09, C.24, supersedes the incorporation of approved Traveller BWR-06, C.8.

124. ITS 5.11

- i. These changes are provided for consistency with the new 10 CFR 20 references. This is an ITS Category (iii) change.

6.0Q44 Document that the NRC staff accepts the new 10 CFR Part 20 Ginna licensing basis.

Response: The new 10 CFR Part 20 was required to be implemented by all licensees by January 1, 1994. The NRC does not have to approve this program for each licensee. Instead, the NRC inspects and audits compliance with the new 10 CFR Part 20. Recent inspections for Ginna include 94-08 and 95-05 (attached).

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Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

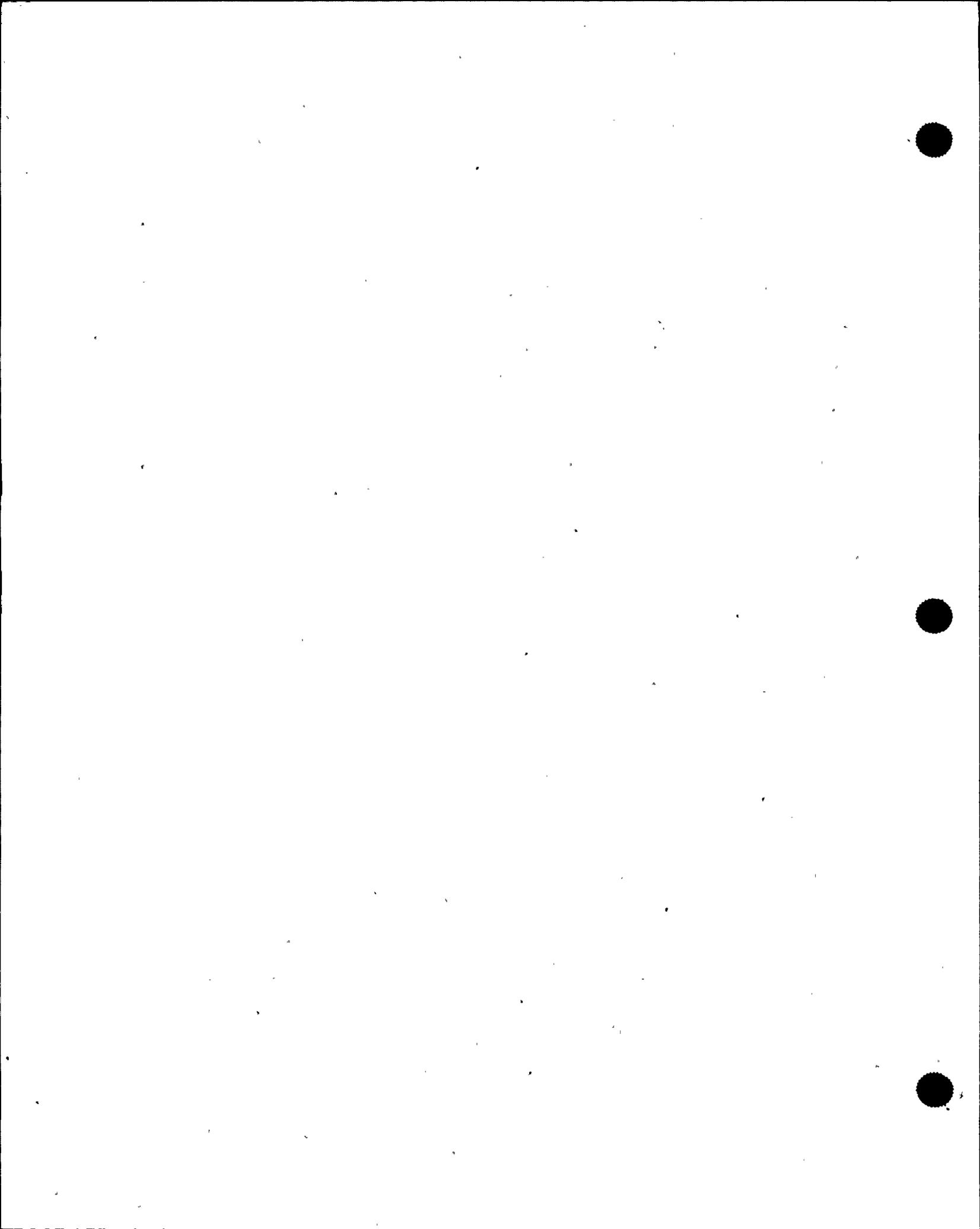
Attachment L
Chapters 1.0 - 3.4

Volume V

Attachment L

"Redlined" Version of Attachment C as Submitted on May 26, 1995

December 1995



1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, time constants display , and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an in-place qualitative assessment of sensor behavior that compares the other sensing elements with the recently installed sensing element.

(158)
(227)

(107)

(continued)

1.1 Definitions (continued)

CHANNEL CALIBRATION
(continued)

The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL OPERATIONAL
TEST (COT)

227

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

CORE ALTERATION

169

CORE ALTERATION shall be the movement of any fuel, sources, or ~~other~~ reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS
REPORT (COLR)

The COLR is the plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

(continued)

1.1 Definitions (continued)

223

The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Revision 1, 1977.

(continued)

1.1 Definitions (continued)

\bar{E} - AVERAGE
DISINTEGRATION ENERGY

\bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

(35)

L_a

~~The maximum allowable primary containment leakage rate, L_a , shall be 0.2% of primary containment air weight per day at the calculated peak containment pressure (P_a).~~

LEAKAGE

LEAKAGE from the RCS shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

(continued)

1.1 Definitions (continued)

~~LEAKAGE~~ ————— c. Pressure Boundary LEAKAGE
~~(continued)~~

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

(continued)

1.1 Definitions (continued)

- MODE** A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
- OPERABLE - OPERABILITY** A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
- PHYSICS TESTS** PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:
- a. Described in Chapter 14, Initial Test Program of the UFSAR;
 - b. Authorized under the provisions of 10 CFR 50.59; or
 - c. Otherwise approved by the Nuclear Regulatory Commission (NRC).

(continued)

1.1 Definitions (continued)

PRESSURE AND
TEMPERATURE LIMITS
REPORT (PTLR)

The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.

QUADRANT POWER TILT
RATIO (QPTR)

QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.

RATED THERMAL POWER
(RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1520 MWt.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and
- b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.

(continued)

1.1 Definitions (continued)

(continued)

1.1 Definitions (continued)

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE
OPERATIONAL TEST
(TADOT)

(227)

A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{off})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Shutdown	< 0.99	NA	≥ 350
4	Hot Standby ^(b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown ^(b)	< 0.99	NA	≤ 200
6	Refueling ^(c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

(169)

BACKGROUND Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, or Frequency.

(228)

EXAMPLES The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1 LOGICAL CONNECTORS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)



1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2 MULTIPLE LOGICAL CONNECTORS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . <u>OR</u> A.2.1 Verify . . . <u>AND</u> A.2.2.1 Reduce . . . <u>OR</u> A.2.2.2 Perform . . . <u>OR</u> A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the plant. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the plant is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the plant is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure with Completion Times based on initial entry into the Condition.

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(continued)

1.3 Completion Times

DESCRIPTION
(continued)

(161)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. ~~The Completion time extension cannot be used to extend the stated Completion Time for the first inoperable train, subsystem, component, or variable.~~ To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

- a. Must exist concurrent with the first inoperability; and
- b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

- a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
- b. The stated Completion Time as measured from discovery of the subsequent inoperability.

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~~The Completion time extension cannot be used to extend the stated Completion Time for the first inoperable train, subsystem, component, or variable.~~

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

(continued)



1.3 Completion Times

DESCRIPTION
(continued)

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry). An example of a modified "time zero" with the Completion Time expressed as "once per 8 hours" is illustrated in Example 1.3-6, Condition A. In this example, the Completion Time may not be extended.

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EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1 COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-2 DEFAULT CONDITIONS/LCO 3.0.3 ENTRY/COMPLETION TIMES

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours, because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One train inoperable.	A.1 Restore train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

When a train is declared inoperable, Condition A is entered. If the train is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable train is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times

EXAMPLES . EXAMPLE 1.3-2 (continued)

When a second train is declared inoperable while the first train is still inoperable, Condition A is not re-entered for the second train. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable train. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if either one of the inoperable trains is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

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While in LCO 3.0.3, if either one of the inoperable trains is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

Upon restoring either one of the trains to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first train was declared inoperable. This Completion Time may be extended if the train restored to OPERABLE status was the first inoperable train. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second train being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3 MULTIPLE FUNCTION COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours
C. One Function X train inoperable. <u>AND</u> One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. <u>OR</u> C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-4 MULTIPLE FUNCTION COMPLETION TIMES

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)



1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5 SEPARATE ENTRY CONDITION

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific condition, the Note would appear in that Condition, rather than at the top of the ACTIONS table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6 MULTIPLE ACTIONS WITHIN A CONDITION/COMPLETION TIME EXTENSIONS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered, and the initial performance of Required Action A.1 must be completed within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A ~~until the LGO is met.~~

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(continued)



1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-7 MULTIPLE ACTIONS WITHIN A CONDITION/COMPLETION TIME EXTENSIONS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE
COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-1 SINGLE FREQUENCY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the plant is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the plant is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the plant is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

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(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-2 MULTIPLE FREQUENCIES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 1.25 times the stated Frequency extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3 FREQUENCY BASED ON SPECIFIED CONDITION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Required to be performed within 12 hours after \geq 25% RTP. -----</p> <p>Perform channel adjustment.</p>	<p>7 days</p>

The interval continues, whether or not the plant operation is $<$ 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is $<$ 25% RTP, this Note allows 12 hours after power reaches \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was $<$ 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power \geq 25% RTP.

Once the plant reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency and the provisions of SR 3.0.3 would apply.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

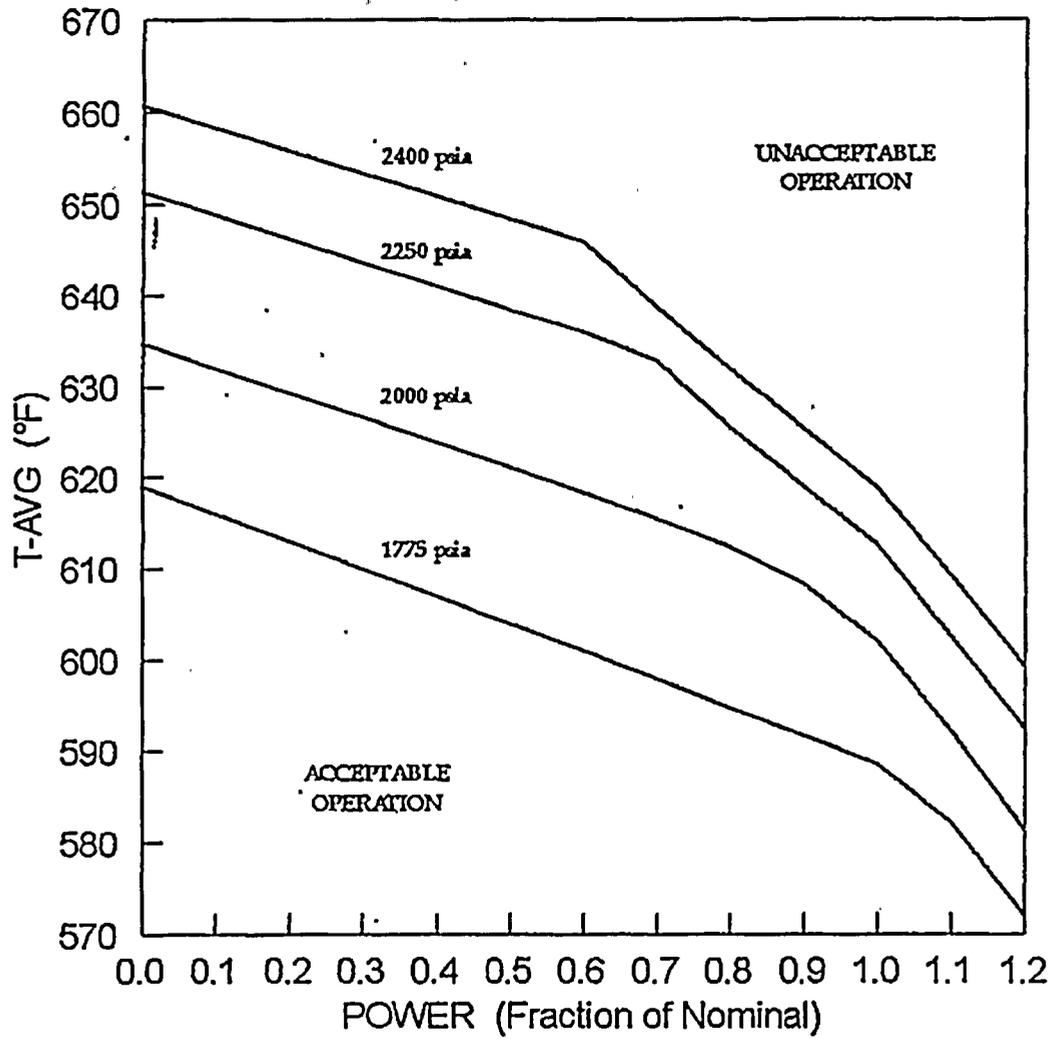


Figure 2.1.1-1
Reactor Safety Limits

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

Atomic Industrial Forum (AIF) GDC 6 (Ref. 1) requires that the reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. This integrity is required during steady state operation, normal operational transients, and anticipated operational occurrences (A00s). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur on the limiting fuel rods and by requiring that fuel centerline temperature stays below the melting temperature (Ref. 2).

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

(continued)

BASES

BACKGROUND
(continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium - water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria (Ref. 3):

- a. The hot fuel pellet in the core must not experience centerline fuel melting; and
- b. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. The effects of these uncertainties have been statistically combined with the correlation uncertainty to determine design limit departure from nucleate boiling ratio (DNBR) values that satisfy the DNB design criterion. The observable parameters, thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and/or WRB-1 DNB correlation. These DNB correlations have been developed to predict the DNB flux and the location of DNB for auxiliary uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. A minimum value of the DNB ratio is specified so that during steady state operation, normal operational transients and anticipated transients, there is a 95% probability at a 95% confidence level that DNB will not occur. The curves of Figure 2.1.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which this minimum DNB value is satisfied. The area of safe operation is at or below these lines.-

Safe operation relative to Figure 2.1.1-1 refers to transient or accident conditions. Normal steady state operation is governed by LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits."

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility (Ref. 4).

The Reactor Trip System setpoints specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation", in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities.

(continued)

BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

Automatic enforcement of these reactor core SLs is provided by the following functions (Ref. 5):

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- a. ~~Overtemperature ΔT High pressurizer pressure~~ trip;
 - b. ~~Overpower ΔT Low pressurizer pressure~~ trip;
 - c. ~~Power Range Neutron Flux Overtemperature ΔT~~ trip; and
 - d. ~~Overpower ΔT~~ trip;
 - e. ~~Power Range Neutron Flux~~ trip; and
 - f. Steam generator safety valves.

Additional ~~anticipatory~~ trip functions are ~~also~~ provided to ~~backup these functions~~ for specific abnormal conditions.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (Ref. 6) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

Figure B 2.1.1-1 shows an example of the reactor core safety limits of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is greater than or equal to the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. From this type of figure, the curves on Figure 2.1.1-1 of the accompanying specification can be generated. Each of the curves of Figure 2.1.1-1 has three distinct slopes. Working from left to right, the first slope ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid such that overtemperature ΔT indication remains valid. The second slope ensures that the hot leg steam quality remains $\leq 15\%$ as required by W-3 correlation. The final slope ensures that DNBR is always ≥ 1.3 .

(continued)



BASES

SAFETY LIMITS
(continued)

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT
VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the plant in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage. If the Completion Time is exceeded, actions shall continue in order to bring the plant to a MODE of operation where this SL is not applicable.

(continued)

BASES

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 6, Issued for comment July 10, 1967.
 2. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Deletion of Information Pertaining to Definition of Hot Channel Factors," dated May 30, 1985.
 3. UFSAR, Section 4.2.1.3.3.
 4. UFSAR, Section 4.4.3.
 5. WCAP-8745, "Design Bases for the Thermal Overpower Delta T and Thermal Overtemperature Delta T Trip Functions," March 1977.
 6. UFSAR, Section 7.2.1.1.1.
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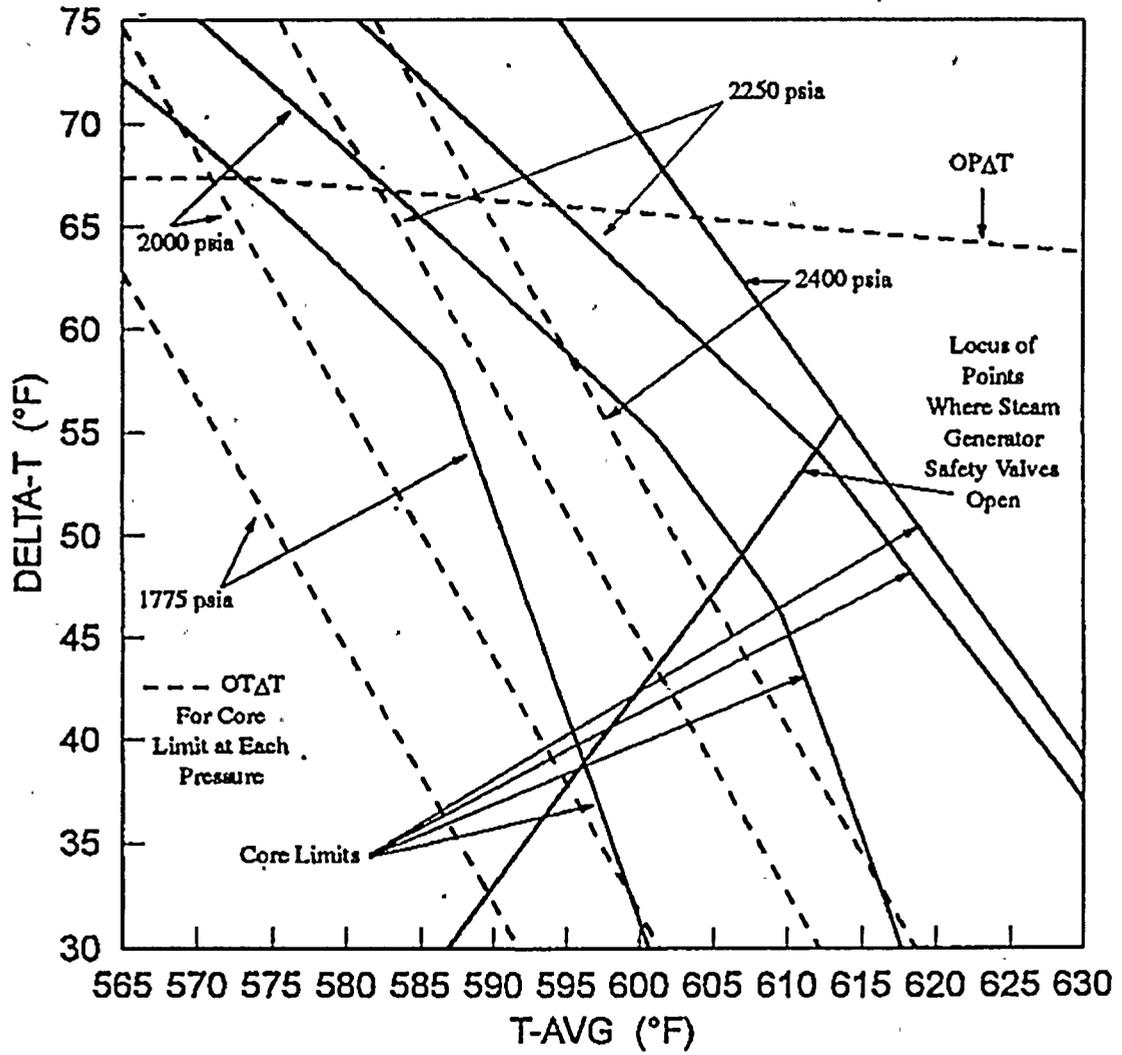


Figure B 2.1.1-1
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to Atomic Industrial Forum (AIF) GDC 9, "Reactor Coolant Pressure Boundary," GDC 33, "Reactor Coolant Pressure Boundary Capability," and GDC 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

The design pressure of the RCS is 2485 psig (Ref. 2). During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 3) except for locked rotor accidents which must be limited to 120% of the design pressure (Refs. 4, 5, and 6). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of plant operation, RCS components are pressure tested, in accordance with the requirements of the approved Ginna ISI/IST Program which is based on ASME Code, Section XI (Ref. 7).

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 8). If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 3) except for locked rotor accidents which must be limited to 120% of the design pressure. The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints, ~~together with the settings of the MSSVs, provide pressure protection for normal operation and AOs.~~ (Ref. ~~9~~), ~~together with the settings of the MSSVs, provide pressure protection against overpressurization (Ref. 9).~~ ~~The reactor high pressure trip setpoint is specifically set to~~ ~~9~~ ~~The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization.~~ The safety analyses which credit either the high pressure trip or the RCS pressurizer safety valves are performed using conservative assumptions relative to the other pressure control devices.

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More specifically, no credit is taken for operation of the following:

- a. Pressurizer power operated relief valves;
- b. Steam generator atmospheric relief valves;
- c. Steam Dump System;
- d. Reactor Control System;
- e. Pressurizer Level Control System; or
- f. Pressurizer spray valves.

(continued)

BASES

SAFETY LIMITS

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The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure except for locked rotor accidents which must be limited to 120% of the design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under the original design requirements for ~~Ginna Station of USAS B31.1~~ (Ref. 5) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

If SL 2.1.2 is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 8).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the Completion Time is exceeded, actions shall continue in order to restore compliance with the SL and bring the plant to MODE 3.

(continued)



BASES

SAFETY LIMIT
VIOLATIONS
(Continued)

If SL 2.1.2 is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. If the Completion Time is exceeded, action shall continue in order to reduce pressure to less than the SL. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 9, 33, and 34, Issued for comment July 10, 1967.
 2. UFSAR, Section 5.1.4.
 3. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
 4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-1, XV-2, XV-3, XV-4, XV-5, XV-6, XV-7, XV-8, XV-10, XV-12, XV-14, XV-15, and XV-17, Design Basis Events, Accidents and Transients (R.E. Ginna)," dated September 4, 1981.
 5. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967 edition.
 6. UFSAR, Section 15.3.2.
 7. ASME, Boiler and Pressure Vessel Code, Section XI.
 8. 10 CFR 100.
 9. UFSAR, Section 7.2.2.2.
-

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and (1) the associated ACTIONS are not met, (2) an associated ACTION is not provided, or (3) if directed by the associated ACTIONS, the plant shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated to place the plant, as applicable, in:

- a. MODE 3 within ~~76~~ hours;
- b. MODE 4 within ~~1312~~ hours; and
- c. MODE 5 within ~~3736~~ hours.

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Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

(continued)



3.0 LCO APPLICABILITY

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to determine OPERABILITY.

(continued)

3.0 LCO APPLICABILITY

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LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.14, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

3.0 LCO APPLICABILITY

164 LCO 3.0.7

Test Exception LCO 3.1.8, "PHYSICS TEST ~~Exception~~Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a SR, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SR APPLICABILITY

SR 3.0.3
(continued) When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the plant is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

(continued)



BASES

LCO 3.0.2
(continued)

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the plant in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the plant that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Condition no longer exists. In this instance, the individual LCO's ACTIONS specify the Required Actions. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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BASES

LCO 3.0.2
(continued)

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems as required by the LCO. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the plant may enter a MODE or other specified condition in which another Specification becomes applicable and the new LCO is not met. In this case, the Completion Times of the new Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or

(continued)

BASES

LCO 3.0.3
(continued)

- b. The condition of the plant is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the plant. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the plant in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

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BASES

LCO 3.0.3
(continued)

Upon entering LCO 3.0.3, the Shift Supervisor shall evaluate the condition of the plant and determine actions to be taken, considering plant safety first, that will allow sufficient time for an orderly plant shutdown. These actions shall include preparation for a safe and controlled shutdown, as well as actions to correct the condition which caused entry into LCO 3.0.3. If it is determined that the condition that caused entry into LCO 3.0.3 can be corrected within a reasonable period of time and still allow sufficient time for an orderly plant shutdown, a power reduction does not have to be initiated. This includes coordinating the reduction in electrical generation with energy operations to ensure the stability and availability of the electrical grid. The shutdown shall be initiated so that the time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A plant shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met.
- b. A Condition exists for which the Required Actions have now been performed.
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

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BASES

LCO 3.0.3

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The time limits of LCO 3.0.3 allow ~~37~~³⁶ hours for the plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next ~~11~~¹⁰ hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of ~~13~~¹² hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the plant is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.11, "Spent Fuel Pool (SFP) Water Level." LCO 3.7.11 has an Applicability of "During movement of irradiated fuel assemblies in the SFP." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.11 are not met while in MODE 1, 2, ~~or~~³, or 4, there is no safety benefit to be gained by placing the plant in a shutdown condition. The Required Action of LCO 3.7.11 of "Suspend movement of irradiated fuel assemblies in the SFP" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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BASES

LCO 3.0.4

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LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the plant in a different MODE or other specified condition stated in ~~that the~~ Applicability when the following exist:

- a. Plant conditions are such that the requirements of an LCO would not be met in the MODE or other specified condition in the Applicability desired to be entered; and
- b. The plant would be required to exit the MODE or other specified condition in the Applicability desired to be entered in order to comply with the Required Actions of the affected LCO.

Compliance with Required Actions that permit continued operation of the plant for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the plant before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

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~~In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a shutdown performed in response to the expected failure to comply with ACTIONS.~~

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~~Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions apply to ACTIONS which allow plant operation in the MODE entry into MODES or other specified condition in the Applicability for only a limited period of time for specified plant conditions in the Applicability when the associated ACTIONS to be entered do~~

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BASES

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not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

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BASES

LCO 3.0.4
(continued)

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits). LCO 3.0.4 is applicable when entering all MODES, as permitted by SR 3.0.1 whether increasing in MODES (e.g., MODE 5 to MODE 4) or decreasing in MODES (e.g., MODE 4 to MODE 5). This requirement precluding entry into another MODE when the associated ACTIONS do not provide for continued operation for an unlimited period of time ensures that the plant maintains sufficient equipment OPERABILITY and redundancy as assumed in the accident analyses.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this LCO is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

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BASES

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

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BASES

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LCO 3.0.5 An example of demonstrating the OPERABILITY of the equipment—
being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

(continued)



BASES

~~LCO 3.0.5~~ ————— An example of demonstrating the OPERABILITY of other
~~(continued)~~ ————— equipment is taking an inoperable channel or trip system out
109 of the tripped condition to prevent the trip function from
occurring during the performance of an SR on another channel
in the other trip system. A similar example of
demonstrating the OPERABILITY of other equipment is taking
an inoperable channel or trip system out of the tripped
condition to permit the logic to function and indicate the
appropriate response during the performance of an SR on
another channel in the same trip system.

LCO 3.0.6 LCO 3.0.6 establishes an exception to LCO 3.0.2 for support
systems that have an LCO specified in the Technical
Specifications (TS). This exception is provided because
LCO 3.0.2 would require that the Conditions and Required
Actions of the associated inoperable supported system LCO be
entered solely due to the inoperability of the support
system. This exception is justified because the actions
that are required to ensure the plant is maintained in a
safe condition are specified in the support systems' LCO's
Required Actions. These Required Actions may include
entering the supported system's Conditions and Required
Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO
specified for it in the TS, the supported system(s) are
required to be declared inoperable if determined to be
inoperable as a result of the support system inoperability.
However, it is not necessary to enter into the supported
systems' Conditions and Required Actions unless directed to
do so by the support system's Required Actions. The
potential confusion and inconsistency of requirements
related to the entry into multiple support and supported
systems' LCO's Conditions and Required Actions are
eliminated by providing all the actions that are necessary
to ensure the plant is maintained in a safe condition in the
support system's Required Actions.

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BASES

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BASES

LCO 3.0.6
(continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.14, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

BASES

LCO 3.0.7

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There are certain special tests and operations required to be performed at various times over the life of the plant. These special tests and operations are necessary to demonstrate select plant performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, "PHYSICS TEST Exception Exceptions - MODE 2," allows specified Technical Specification (TS) requirements to be changed to permit performances of special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the Test Exception LCO is used as an allowable exception to the requirements of a Specification.

~~Unplanned events may satisfy the requirements for a given SR surveillances, including surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance~~

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(continued)

BASES

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~~includes those SRs whose performance is normally precluded
in a given MODE or other specified conditions.~~

(continued)

BASES

~~SR 3.0.1~~ ~~Surveillances, including Surveillances invoked by Required~~
~~(continued)~~ ~~Actions, do not have to be performed on inoperable equipment~~
~~because the ACTIONS define the remedial measures that apply.~~
232 Surveillances have to be met and performed in accordance
with SR 3.0.2, prior to returning equipment to OPERABLE
status.

(continued)

BASES

(continued)

~~SR 3.0.1~~ Upon completion of maintenance, appropriate post maintenance-
testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary plant parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

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~~If subsequent post maintenance testing fails, the appropriate component LCO shall be entered.~~

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

(continued)



BASES

SR 3.0.2
(continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. ~~An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions."~~ The requirements of regulations take precedence over the TS. ~~The TS cannot in and of themselves extend a test interval.~~ Therefore, when a test interval is specified in the regulations, the test interval cannot be exceeded by TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable."

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~~Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."~~ "An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with Refueling intervals) or periodic Completion Time intervals beyond those specified.

(continued)

BASES

(continued)

BASES

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified plant conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

BASES

SR 3.0.3
(continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with plant shutdown as well as startup.

The provisions of this specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from a shutdown performed in response to the expected failure to comply with ACTIONS.

(continued)

BASES

SR 3.0.4
(continued)

However, in certain circumstances failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, train, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

Se LCO 3.0.4 is applicable when entering all MODES, whether increasing in MODES (e.g., MODE 5 to MODE 4) or decreasing in MODES (e.g., MODE 4 to MODE 5). This requirement precluding entry into another MODE when the associated ACTIONS do not provide for continued operation for an unlimited period of time ensures that the plant maintains sufficient equipment OPERABILITY and redundancy as assumed in the accident analyses.

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits specified in the COLR.

APPLICABILITY: MODE 2 with $k_{off} < 1.0$,
MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is within the limits specified in the COLR.	48 ²⁴ hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within $\pm 1\% \Delta k/k$ of predicted values.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	<u>AND</u> A.2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.1</p> <p>(170) -----NOTE----- Only required Required to be performed prior to entry in entering MODE 1.</p> <p>Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p>Once after each refueling</p>
<p>SR 3.1.2.2</p> <p>(169) -----NOTES-----</p> <ol style="list-style-type: none"> 1. Only required after 60 effective full power days (EFPD). 2. The predicted reactivity values may must be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. <p>Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p>31 EFPD</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3

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The MTC shall be maintained within the ~~beginning of cycle life (BOL) limit and the end of cycle life (EOL) limit~~ limits specified in the COLR. The maximum upper limit shall be less than or equal to 5 pcm/°F for power levels below 70% RTP and less than or equal to 0 pcm/°F at or above 70% RTP.

APPLICABILITY: MODE 1 and MODE 2 with $k_{off} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. NOTE Required Action A.1 must be completed whenever Condition A is entered.</p> <p>MTC not within upper limit.</p>	<p>A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.</p>	<p>24 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 2 with $k_{off} < 1.0$.</p>	<p>6 hours</p>

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>57</p> <p>C. -----NOTE----- Required Action C.1 must be completed whenever Condition C is entered. -----</p> <p>55</p> <p>Projected EOL end of cycle life (EOL) MTC not within lower limit.</p>	<p>-----NOTE----- ECO 3.0.4 is not applicable. -----</p> <p>C.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.</p>	<p>Once prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm</p>
<p>D. Required Action and associated Completion Time of Condition C not met.</p>	<p>D.1 Be in MODE 4.</p>	<p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.3.1 Verify MTC is within upper limit.</p>	<p>Once prior to entering MODE 1 after each refueling</p>
<p>55</p> <p>SR 3.1.3.2 Confirm that MTC will be within limits at 70% RTP and at EOL.</p>	<p>Once prior to entering MODE 1 after each refueling</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.3.3 Confirm that MTC will be within limits at EOL.	Once prior to entering MODE 1 after each refueling.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODE 1,
MODE 2 with $K_{off} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>THE FOLLOWING TEXT WAS MOVED A.A. THE PRECEDING TEXT WAS MOVED One or more rod(s) untrippable.</p>	<p>A.1.1 Verify SDM is within the limits specified in the COLR.</p> <p><u>OR</u></p> <p>A.1.2 Initiate boration to restore SDM to within limit.</p> <p><u>AND</u></p> <p>A.2 Be in MODE 2 with $K_{off} < 1.0$.</p>	<p>1 hour</p> <p>1 hour</p> <p>6 hours</p>
<p>B. One rod not within alignment limits.</p>	<p>B.1.1 Verify SDM is within the limits specified in the COLR.</p> <p><u>OR</u></p> <p>B.1.2 Initiate boration to restore SDM to within limit.</p> <p><u>AND</u></p>	<p>1 hour</p> <p>1 hour</p> <p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce THERMAL POWER to $\leq 75\%$ RTP.	2 hours
	<u>AND</u>	
	B.3 Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>	
	B.4 Perform SR 3.2.1.1.	72 hours
	<u>AND</u>	
	B.5 Perform SR 3.2.2.1.	72 hours
	<u>AND</u>	
	B.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 2 with $K_{off} < 1.0$.	6 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	D:1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 2 with $K_{\text{eff}} < 1.0$.	6 hours

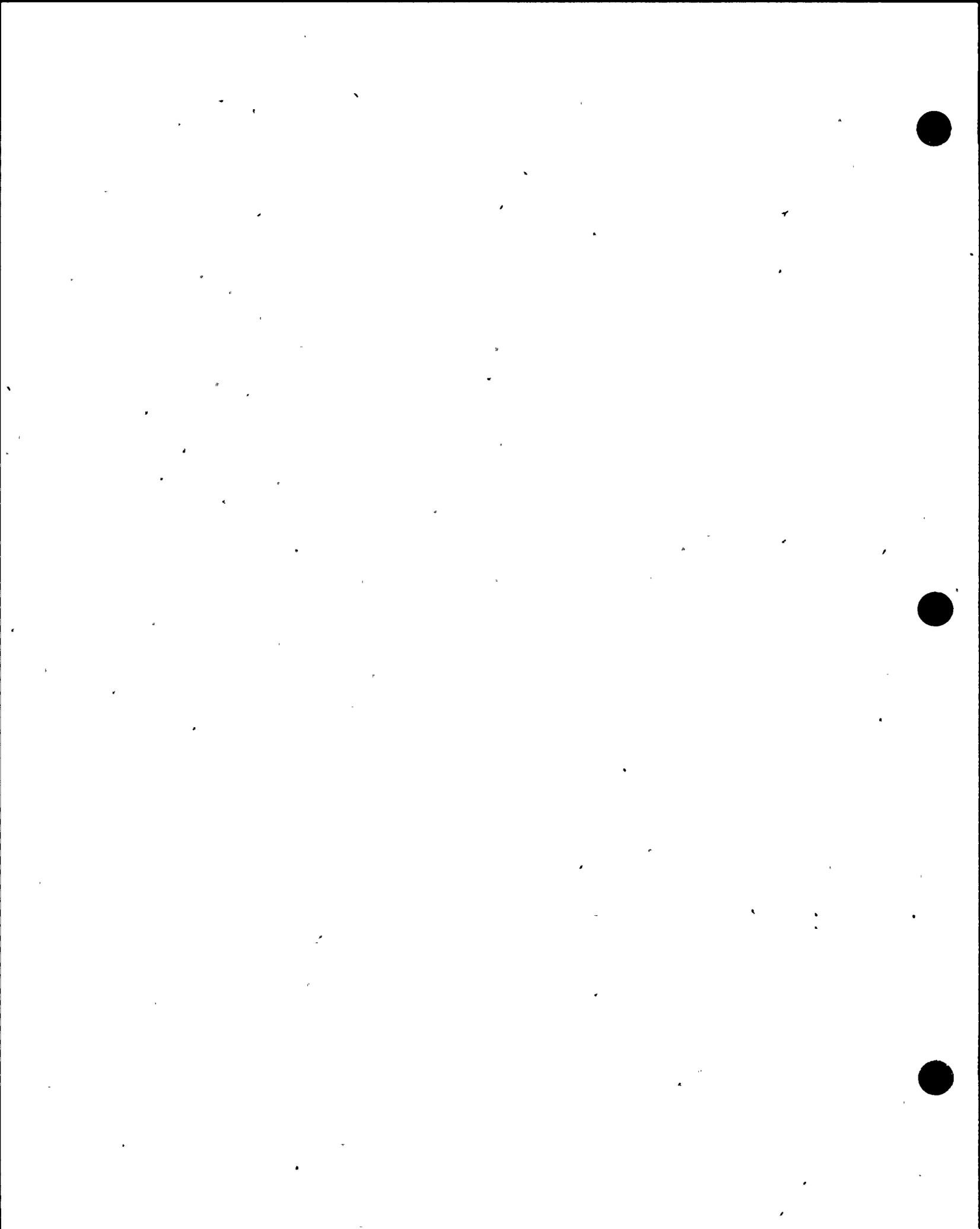
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2 -----NOTE----- Only required to be performed if the rod position deviation monitor is inoperable. ----- Verify individual rod positions within alignment limit.	Once within 4 hours and every 4 hours thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.3 Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core to a MRPI transition in either direction.</p>	<p>92 days</p>
<p>SR 3.1.4.4 Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with:</p> <ul style="list-style-type: none"> a. $T_{avg} \geq 500^{\circ}F$; and b. Both reactor coolant pumps operating. 	<p>Once prior to reactor criticality after each removal of the reactor head</p>



3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limit

LCO 3.1.5 The shutdown bank shall be at or above the insertion limit specified in the COLR.

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-----NOTE-----
The shutdown bank may be outside the limit when required for performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
 MODE 2 with $K_{off} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shutdown bank not within limit.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown bank to within limit.	2 hours
B. Required Action and associated Completion Time not met. B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{off} < 1.0$.	6 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify the shutdown bank insertion is within the limit specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

-----NOTE-----
The control bank being tested may be outside the limits when required for the performance of SR 3.1.4.3.

APPLICABILITY: MODE 1,
MODE 2 with $k_{off} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank limits not met.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore control bank(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2 with $K_{off} < 1.0$.	6 hours
B. Required Action and associated Completion Time not met.		

ACTIONS (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.6.1 Verify estimated critical control bank position is within the limits specified in the COLR.</p>	<p>Once within 4 hours prior to achieving criticality</p>
<p>SR 3.1.6.2 Verify each control bank insertion is within the limits specified in the COLR.</p>	<p>12 hours</p>
<p>SR 3.1.6.3 -----NOTE----- Only required to be performed if the rod insertion limit monitor is inoperable. ----- Verify each control bank insertion is within the limits specified in the COLR.</p>	<p>Once within 4 hours and every 4 hours thereafter</p>
<p>SR 3.1.6.4 Verify each control bank not fully withdrawn from the core is within the sequence and overlap limits specified in the COLR.</p>	<p>12 hours</p>

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3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LC0 3.1.7 The Microprocessor Rod Position Indication (MRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODE 2 with $K_{eff} \geq 1.0$.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each group with no more than one inoperable rod position indicator in the inoperable MRPI per group and for each bank with no more than one inoperable each demand position indicator in the per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to $\leq 50\%$ RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. More than one MRPI per group inoperable for one or more groups.</p> <p><u>OR</u></p> <p>More than one demand position indicator per bank inoperable for one or more banks.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.7.1 Verify each MRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.</p>	<p>Prior to reactor criticality after each removal of the reactor head</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

- LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.4, "Rod Group Alignment Limits";
- LCO 3.1.5, "Shutdown Bank Insertion Limit";
- LCO 3.1.6, "Control Bank Insertion Limits";
- LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

- a. THERMAL POWER is maintained \leq 5% RTP;
- b. RCS lowest loop average temperature is \geq 530°F; and
- c. SDM is within the limits specified in the COLR.

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APPLICABILITY: ~~MODE 2 during~~ During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u> A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Perform a COT on power range and intermediate range channels per SR 3.3.1.7 and SR 3.3.1.8.	Once within 7 days prior to criticality
SR 3.1.8.2 Verify the RCS lowest loop average temperature is $\geq 530^{\circ}\text{F}$.	30 minutes
SR 3.1.8.3 Verify THERMAL POWER is $\leq 5\%$ RTP.	30 minutes
SR 3.1.8.4 Verify SDM is within the limits specified in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 27 and 28 (Ref. 1), two independent reactivity control systems must be available and capable of holding the reactor core subcritical from any hot standby or hot operating condition. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs) which are defined as Condition 2 events in Reference 2 (i.e., events which can be expected to occur during a calendar year with moderate frequency). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn and the fuel and moderator temperature are changed to the nominal hot zero power temperature.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable rod cluster control assemblies (RCCAs) and soluble boric acid in the Reactor Coolant System (RCS) which each provide a neutron absorbing mechanism. The Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The chemical and volume control system can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

(continued)

BASES

BACKGROUND
(continued)

During power operation, SDM control is ensured by operating with the shutdown bank fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank fully withdrawn position is defined in the COLR. When the plant is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE
SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in the safety analyses. The safety analysis (Ref. 3) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out following a scram.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Accidents;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 200 cal/gm energy deposition for the rod ejection accident); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The most limiting accident for the SDM requirements is based on a steam line break (SLB), as described in the accident analysis (Ref. 3). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting SLB for both one loop and two loop operation, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the SLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting SLB transient, the SDM requirement must also protect against:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition;
- c. Startup of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

Each of these events is discussed below.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

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In the boron dilution analysis (Ref. 4), the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis (i.e., the time available to operators to stop the dilution event). This event is analyzed for refueling, shutdown (MODE 5) and power operation conditions and is most limiting at the beginning of core life, shutdown and power operation conditions and is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip (Ref. 5). In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits if SDM has been maintained.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core (Ref. 6). The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less severe than the effects of a small steam line break with one loop operation. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition if SDM has been maintained.

The ejection of a control rod constitutes a break in the RCS which rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure (Ref. 7). The ejection of a rod also produces a time dependent redistribution of core power which results in a high neutron flux trip. Fuel and cladding limits are not exceeded if SDM has been maintained.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the plant is operating within the bounds of accident analysis assumptions.

(continued)



BASES

LCO . SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration in the RCS.

The COLR provides the shutdown margin requirement with respect to RCS boron concentration. The SLB (Ref. 3) and the boron dilution (Ref. 4) accidents are the most limiting analyses that establish the SDM curve in the COLR. The maximum shutdown margin requirement occurs at end of cycle life and is based on the value used in analysis for the SLB. Early in cycle life, less SDM is required and is bounded by the requirements provided in the COLR. All other accidents analyses are based on 1% reactivity shutdown margin. For SLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 8). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

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APPLICABILITY

In MODE 2 with $k_{off} < 1.0$ and in MODES 3, 4 and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODE 1 and MODE 2 with $K_{off} \geq 1.0$, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits."

(continued)

BASES

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the flowpath of choice would utilize a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 10 gpm using 13,000 ppm boric acid solution, ~~Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 10 gpm,~~ it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 10 gpm and 13,000 ppm represent typical values and are provided for the purpose of offering a specific example.

(109)

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.1.1

In MODE 2 with $K_{\text{eff}} < 1.0$ and MODES 3, 4, and 5, the SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

64
SDM. The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. The Frequency of 48 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27 and 28, Issued for comment July 10, 1967.
2. "American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
3. UFSAR, Section 15.1.5.
4. UFSAR, Section 15.4.4.
5. UFSAR, Section 15.4.2.
6. UFSAR, Section 15.4.3.
7. UFSAR, Section 15.4.5.
8. 10 CFR 100.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 30 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN (SDM) or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" in ensuring the reactor can be brought safely to cold; ~~"SHUTDOWN MARGIN (SDM)" in ensuring the reactor can be brought safely to cold,~~ subcritical conditions.

(149)

(continued)

BASES

BACKGROUND
(continued)

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve) in the core design report, which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed or stable (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and normal operating temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant moderator temperature. ~~the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER.~~ The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

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(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the Nuclear Design Methodology provides an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOL, or that an unexpected change in core conditions has occurred.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of $\pm 1\% \Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

(continued)

BASES

APPLICABILITY The limits on core reactivity must be maintained during MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODE 2 with $K_{eff} < 1.0$ or MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is only changing because of xenon.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (SR 3.1.2.1).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{\text{eff}} < 1.0$ within 6 hours. If the SDM for MODE 2 with $K_{\text{eff}} < 1.0$ is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{\text{eff}} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

Core reactivity must be verified following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The comparison must be made prior to entering MODE 1 when the core conditions such as control rod position, moderator temperature, and samarium concentration are fixed or stable. Since the reactor must be critical to verify core reactivity, it is acceptable to enter MODE 2 with $K_{eff} \geq 1.0$ to perform this SR. This SR is modified by a Note to clarify that the SR does not need to be performed until prior to entering MODE 1.

SR 3.1.2.2

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Frequency of 31 EFPD, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly. The SR is modified by two Notes. The first Note states that the SR is only required after 60 effective full power days (EFPD). The second Note indicates that the normalization of predicted core reactivity to the measured value must take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 30, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to Atomic Industrial Forum (AIF) GDC 8 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). MTC is defined as the change in reactivity per degree change in moderator temperature since temperature is directly proportional to coolant density. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle life (BOL) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOL within the range analyzed in the plant accident analysis. The end of cycle life (EOL) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOL limit.

(continued)

BASES

BACKGROUND
(continued)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses.

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The UFSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive (i.e., upper limit). Such accidents include the rod withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative (i.e., lower limit). Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is at BOL or EOL. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

MTC satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within the specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

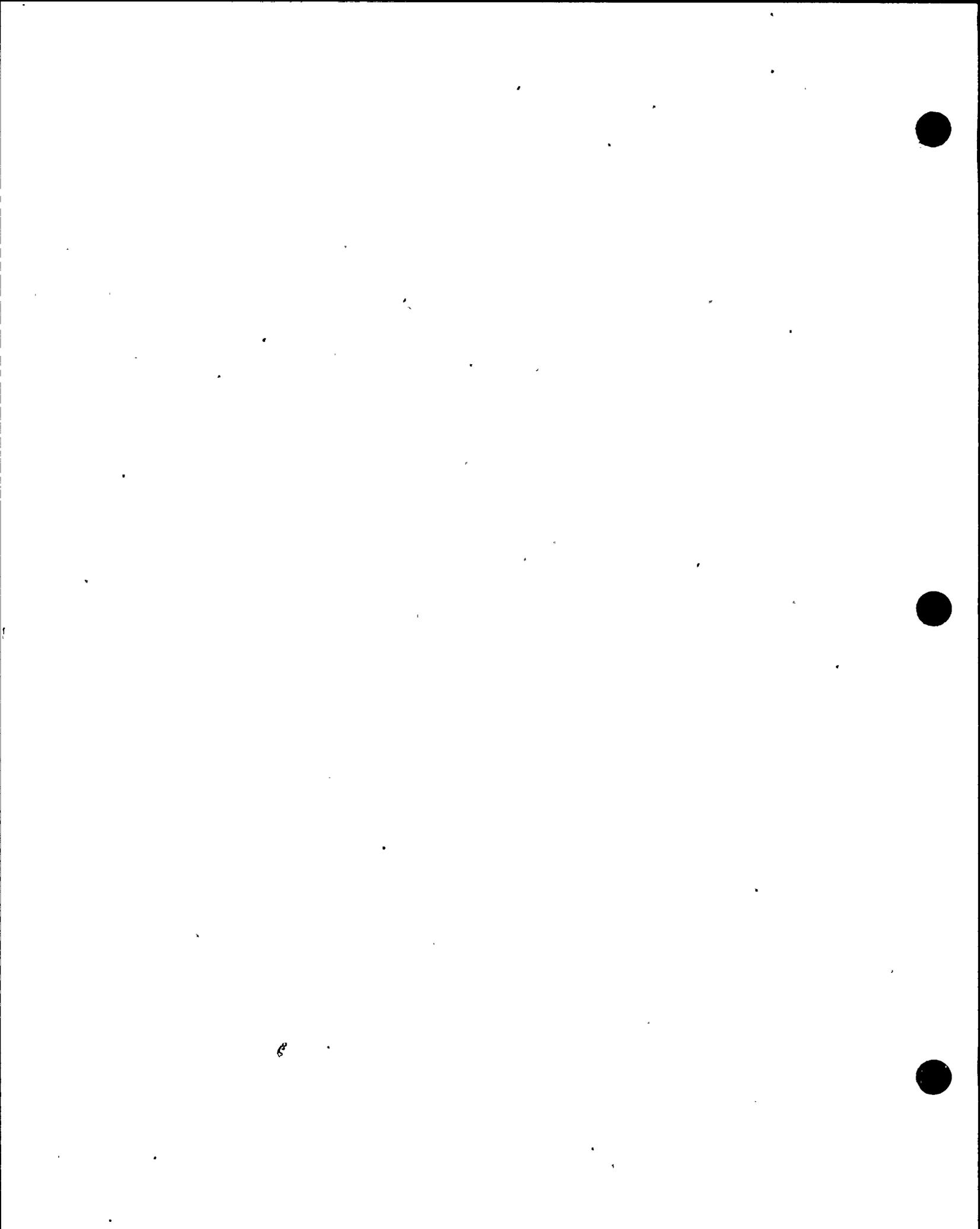
Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at BOL; this upper bound must not be exceeded. This maximum upper limit occurs at BOL, all rods out (ARO), hot zero power (HZP) conditions. At EOL the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance check at BOL on MTC provides confirmation that the MTC is behaving as anticipated and will be within limits at 70% RTP, full power, and EOL so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOL positive limit and the EOL negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the plant to take advantage of improved fuel management and changes in plant operating schedule.

If the LCO limits are not met, the plant response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

(continued)



BASES

APPLICABILITY

In MODE 1, the upper and lower limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis since MTC becomes more negative as the cycle burnup increases because the RCS boron concentration is reduced. ~~to ensure that cooldown accidents will not violate the assumptions of the accident analysis since MTC becomes more negative as the cycle burnup increases, the RCS boron concentration is reduced.~~ In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

(109)

ACTIONS

A.1

MTC must be kept within the upper limit specified in LCO 3.1.3 to ensure that assumptions made in the safety analysis remain valid. The upper limit of Condition A is the upper limit specified in the COLR since this value will always be less than or equal to the maximum upper limit specified in the LCO.

If the upper MTC limit is violated at BOL, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits in the future. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The plant is no

(continued)

BASES

longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

(continued)

BASES

ACTIONS ~~B.1~~
(continued)

ACTIONS ~~A.1~~ (continued)

(5)

~~Condition A has been modified by a Note that requires that Required Action A.1 must be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate where MTC would be above the upper limit specified in the COLR.~~

B.1

If the required administrative withdrawal limits at BOL are not established within 24 hours, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status the plant must be brought to MODE 2 with $k_{\text{eff}} < 1.0$. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOL MTC lower limit means that the safety analysis assumptions of the EOL accidents that use a bounding negative MTC value may be invalid. If it is determined during physics testing that the EOL MTC value will exceed the most negative MTC limit specified in the COLR, the safety analysis and core design must be re-evaluated prior to reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm to ensure that operation near the EOL remains acceptable. The 300 ppm limit is sufficient to prevent EOL operation at or below the accident analysis MTC assumptions.

Condition C has been modified by a Note that requires that Required Action C.1 must be completed whenever this Condition is entered. This is necessary to ensure that the plant does not operate at conditions where the MTC would be below the most negative limit specified in the COLR.

(continued)



BASES

(5)

Required Action C.1 is modified by a Note which states that LCO 3.0.4 is not applicable. This Note is provided since the requirement to re-evaluate the core design and safety analysis prior to reaching an equivalent RTP ARO boron concentration of 300 ppm is adequate action without restricting entry into MODE 1.

(continued)



BASES

ACTIONS
(continued)

D.1

If the re-evaluation of the accident analysis cannot support the predicted EOL MTC lower limit, or if the Required Actions of Condition C are not completed within the associated Completion Time the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 4 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.1

This SR requires measurement of the MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOL MTC value for ARO will be inferred from isothermal temperature coefficient (ITC) measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOL MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

The measurement of the MTC at the beginning of the fuel cycle is adequate to confirm that the MTC remains within its upper limits and will be within limits at 70% RTP, full power and at EOL, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup. This measurement is consistent with the recommendations detailed in Reference 4.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.3.2

(continued)

This SR requires measurement of MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the 70% RTP MTC limit. The frequency of "once prior to MODE 1 after each refueling" ensures the limit will also be met at higher power levels.

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SR 3.1.3.3

This SR requires measurement of MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the most negative MTC LCO. Meeting this limit prior to entering MODE 1 ensures that the limit will also be met at EOL.

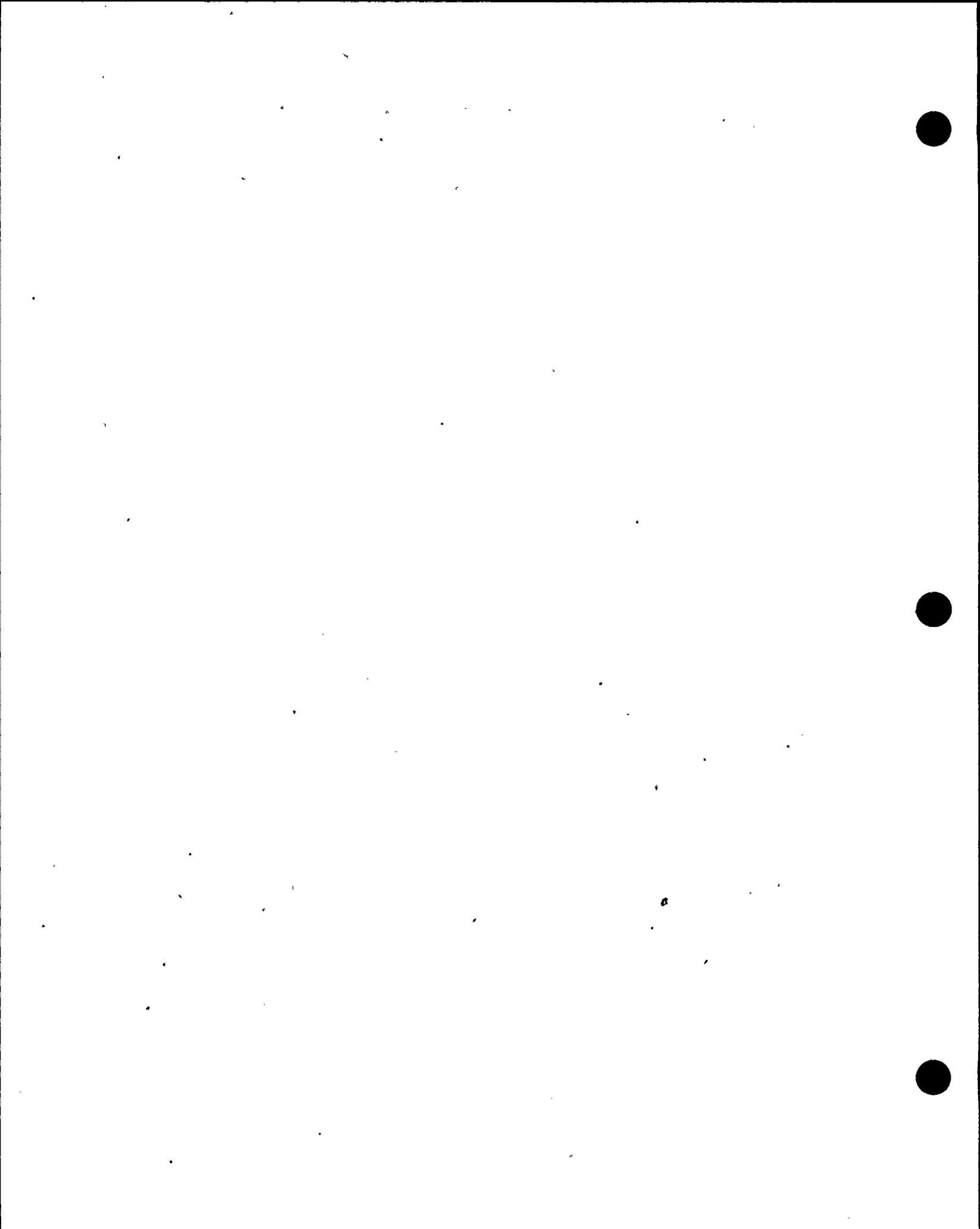
(continued)

~~This SR requires measurement of MTC at BOL prior to entering MODE 1 in order to demonstrate compliance with the 70% RTP MTC limit and the most negative MTC LCO. Meeting these limits prior to entering MODE 1 ensures that the limit will also be met at higher power levels and at EOL.~~

The MTC value for EOL is also inferred from the ITC measurements. The EOL value is calculated using the predicted EOL MTC from the core design report and the difference between the measured and predicted ITC. The EOL value is directly compared to the most negative EOL value established in the COLR to ensure that the predicted EOL negative MTC value is within the accident analysis assumptions.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 8, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
 3. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
 4. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06", dated April 28, 1988.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 6, 14, 27 and 28 (Ref. 1), and 10 CFR 50.46 (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are movable neutron absorbing devices which are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately $\frac{1}{8}$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(continued)

BASES

BACKGROUND
(continued)

The RCCAs are divided among control banks and a shutdown bank. Control banks are used to compensate for changes in reactivity due to variations in operating conditions of the reactor such as coolant temperature, power level, boron or xenon concentration. The shutdown bank provides additional shutdown reactivity such that the total shutdown worth of the bank is adequate to provide shutdown for all operating and hot zero power conditions with the single RCCA of highest reactivity worth fully withdrawn. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station.

The shutdown bank is maintained either in the fully inserted or fully withdrawn position. The fully withdrawn position is defined in the COLR. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Microprocessor Rod Position Indication (MRPI) System.

(continued)

BASES

BACKGROUND
(continued)

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{1}{8}$ inch), but if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The MRPI System also provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The MRPI system consists of one digital detector assembly per rod. All the detector assemblies consist of one coil stack which is multiplexed and becomes input to two redundant MRPI signal processors. Each signal processor independently monitors all rods and senses a rod bottom for any rod. The MRPI system directly senses rod position in intervals of 12 steps for each rod. The digital detector assemblies consist of 20 discrete coil pairs spaced at 12-step intervals. The true rod position is always within ± 8 steps of the indicated position (± 6 steps due to the 12-step interval and ± 2 steps transition uncertainty due to processing and coil sensitivity). With an indicated deviation of 12 steps between the group step counter and MRPI, the maximum deviation between actual rod position and the demand position would be 20 steps, or 12.5 inches.

The safety concerns associated with the MRPI system are associated with generation of a rod drop/rod stop signal which blocks auto rod withdrawal and the ability to comply with the rod misalignment requirement. A rod bottom signal from both signal processors is required to generate a rod drop/rod stop signal. The two-out-of-two coincident signal requirement reduces inadvertent rod drop/rod stop but does not affect the accident analysis assumptions.

The bank demand position and the MRPI rod position signals are monitored by a rod deviation monitoring system that provides an alarm whenever the individual rod position signal deviates from the bank demand signal by > 12 steps. The rod deviation alarm will be generated by the Plant Process Computer System (PPCS).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue (i.e., static rod misalignment). This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Three types of analysis are performed in regard to static rod misalignment (Ref. 4). The first type of analysis considers the case where any one rod is completely inserted into the core with all other rods completely withdrawn. With control banks at their insertion limits, the second type of analysis considers the case when any one rod is completely inserted into the core. The third type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in all three of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

The second type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA fully withdrawn following a main steam line break (Ref. 5).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F_{\Delta H}^N$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

All shutdown and control rods must be OPERABLE to provide the negative reactivity necessary to provide adequate shutdown for all operating and hot zero power conditions. Shutdown and control rod OPERABILITY is defined as being trippable such that the necessary negative reactivity assumed in the accident analysis is available. If a control rod(s) is discovered to be immovable but remains trippable and aligned, the control rod is considered to be OPERABLE.

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

(continued)

BASES

LCO
(continued)

The requirement to maintain the rod alignment of each individual rod position as indicated by MRPI to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis with respect to power distribution and SDM is 25 steps, while a total misalignment from fully withdrawn to fully inserted is assumed for the control rod misalignment accident.

The rod position deviation monitor is used to verify rod alignment on a continuous basis and will provide an alarm whenever the individual rod position signal deviates from the bank demand signal by > 12 steps. Verification that the rod positions are within the alignment limit is made every 12 hours (SR 3.1.4.1). When the rod position deviation monitor is inoperable a verification that the rod positions are within limit must be made more frequently (SR 3.1.4.2).

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODE 1 and MODE 2 with $K_{off} \geq 1.0$ because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODE 2 with $K_{off} < 1.0$ and MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODE 2 with $K_{off} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

(continued)

BASES

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM. Boration is assumed to continue until the required SDM is restored.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a remaining rod of maximum worth.

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{off} < 1.0$ within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{off} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

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When a rod is misaligned, it can usually be moved and is still trippable. If the rod cannot be realigned within 1 hour, then SDM must be verified to be within the limits specified in the COLR or boration must be initiated to restore the SDM. The Completion Time of 1 hour gives the operator sufficient time to perform either action in an orderly manner. 2, B.3, B.4, B.5, and B.6

(continued)

BASES

ACTIONS B.2, B.3, B.4, B.5, and B.6

~~(continued)~~
~~6 (continued)~~

~~Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref.~~

~~THE FOLLOWING TEXT WAS MOVED~~

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

~~Reduction of power to \leq 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref.~~

~~THE PRECEDING TEXT WAS MOVED~~

6). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F_{\Delta H}^N$ are within the required limits (i.e., SR 3.2.1.1 and SR 3.2.2.1) ensures that current operation at \leq 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. ~~ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power.~~ The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Accident for the duration of operation under these conditions. A Completion Time of 5 days is sufficient

(continued)

BASES

time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS C.1
(continued)

C.1

When Required Actions of Condition B cannot be completed within their Completion Time, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 2 with $K_{eff} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

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~~D.0 from full power conditions in an orderly manner and without challenging the plant systems-1.1 and D.1.2~~

~~More than one control rod becoming misaligned from its group position is not expected, (continued)~~

~~More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration is assumed to continue until the required SDM is restored.~~

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the plant conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the plant must be brought to a MODE or Condition in which the LCO requirements are not applicable.

(continued)

BASES

To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours.

~~THE FOLLOWING TEXT WAS MOVED~~

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(continued)



BASES

ACTIONS D.2 (continued)

~~THE PRECEDING TEXT WAS MOVED~~

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 with $K_{off} < 1.0$ from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits using MRPI or the PPCS at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

When the rod position deviation monitor (i.e., the PPCS) is inoperable, no control room alarm is available between the normal 12 hour Frequency to alert the operators of a rod misalignment. A reduction of the Frequency to 4 hours provides sufficient monitoring of the rod positions when the monitor is inoperable. This Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

This SR is modified by a Note that states that performance of this SR is only necessary when the rod position deviation monitor is inoperable.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.4.3

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2 with $K_{eff} \geq 1.0$, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod to a MRPI transition will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. During or between required performances of SR 3.1.4.3 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.4

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. ~~This testing is performed with both RCPs operating and the average moderator temperature $\geq 500^\circ\text{F}$ to simulate a reactor trip under actual conditions. This testing is performed with all RCPs operating and the average moderator temperature $\geq 500^\circ\text{F}$ to simulate a reactor trip under actual conditions.~~

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This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the

(continued)

BASES

potential for an unplanned plant transient if the
Surveillance were performed with the reactor at power.

(continued)

BASES

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 6, 14, 27, and 28, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.4.6.
 5. UFSAR, Section 15.1.5.
 6. UFSAR, Section 15.4.2.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limit

BASES

BACKGROUND

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and a shutdown bank. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The shutdown bank insertion limit is defined in the COLR. The shutdown bank is required to be at or above the insertion limit lines.

(continued)

BASES

BACKGROUND
(continued)

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The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating or diluting). They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity change associated with large changes in RCS temperature.

The design calculations are performed with the assumption that the shutdown bank is withdrawn first. The shutdown bank can be fully withdrawn without the core going critical. The fully withdrawn position is defined in the COLR. This provides available negative reactivity in the event of boration errors. The shutdown bank is controlled manually by the control room operator. The shutdown bank is either fully withdrawn or fully inserted. The shutdown bank must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown bank is then left in this position until the reactor is shut down. The shutdown bank affects core power and burnup distribution, and adds negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

(continued)

BASES

BACKGROUND
(continued)

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The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the shutdown and control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and ensure the required SDM is maintained. ~~the control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.~~

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE
SAFETY ANALYSES.

On a reactor trip, all RCCAs (shutdown bank and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown bank shall be at or above the insertion limit and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and the shutdown bank (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment is that:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. There be no violations of:
 - 1. Specified acceptable fuel design limits, or
 - 2. RCS pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limit affects safety analysis involving core reactivity and SDM (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Refs. 4, 5, 6, and 7).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits, together with AFD, QPTR and the control and shutdown bank alignment limits, ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Refs. 4, 5, 6, and 7).

The shutdown bank insertion limit preserves an initial condition assumed in the safety analyses and, as such, satisfies Criterion 2 of the NRC Policy Statement.

5? LCO The shutdown bank must be at or above the insertion limit any time the reactor is critical and prior to withdrawal of any control rod. LCO The shutdown bank must be at or above the insertion limit any time the reactor is critical. This ensures that a sufficient amount of negative

(continued)

BASES

reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

(continued)

BASES

LCO
(continued)

The LCO is modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

The shutdown bank insertion limit is defined in the COLR.

APPLICABILITY

The shutdown bank must be within the insertion limit, with the reactor in MODE 1 and MODE 2 with $K_{\text{eff}} \geq 1.0$. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 2 with $K_{\text{eff}} < 1.0$ and MODE 3, 4, 5, or 6, the shutdown bank insertion limit does not apply because the reactor is shutdown and not producing fission power. In shutdown MODES the OPERABILITY of the shutdown rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. Refer to LCO 3.1.1 for SDM requirements in MODE 2 with $K_{\text{eff}} < 1.0$ and MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

A.1.1, A.1.2, and A.2

When the shutdown bank is not within insertion limit, verification of SDM or initiation of boration to regain SDM within 1 hour is required, since the SDM in MODE 1 and MODE 2 with $K_{\text{eff}} \geq 1.0$ is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). If the shutdown bank is not within the insertion limit, then SDM will be verified by performing a reactivity balance calculation, taking into account RCS boron concentration, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. Two hours is allowed to restore the shutdown bank to within the insertion limit. This time limit is necessary because the available SDM may be significantly reduced, with the shutdown bank not within the insertion limit. The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to a MODE where the LCO is not applicable. To achieve this status, the plant must be placed in MODE 2 with $k_{\text{eff}} < 1.0$ within a Completion Time of 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1

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Since the shutdown bank is positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of every 12 hours is adequate to ensure that the bank is within the insertion limit. ~~a verification of shutdown bank position at a Frequency of every 12 hours, is adequate to ensure that the bank is within the insertion limit.~~ Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

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BASES

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.1.5.
 5. UFSAR, Section 15.4.1.
 6. UFSAR, Section 15.4.2.
 7. UFSAR, Section 15.4.6.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

(149)

The insertion limits of the shutdown and control rods define the deepest insertion into the core with respect to core power which is allowed and are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SHUTDOWN MARGIN (SDM). The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth SHUTDOWN MARGIN (SDM), and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32 (Ref. 1), and 10 CFR 50.46 (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and a shutdown bank. Each bank is further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The insertion limits figure in the COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks..

(continued)

BASES

BACKGROUND
(continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating or diluting). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. The fully withdrawn position is defined in the COLR. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature.

The rod insertion limit monitor is used to verify control rod insertion on a continuous basis and will provide an alarm whenever the control bank insertion deviates from the rod insertion limits specified in the COLR. Verification that the control banks are within the insertion limit is made every 12 hours (SR 3.1.6.2). When the rod insertion limit monitor is inoperable a verification that the rod positions are within the limit must be made more frequently (SR 3.1.6.3).

The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the fully withdrawn position, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is near the fully withdrawn position at RTP. The insertion sequence is the opposite of the withdrawal sequence (i.e., bank D is inserted first) but follows the same overlap pattern. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

(continued)



BASES

BACKGROUND
(continued)

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limit," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the shutdown and control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and ensure the required SDM is maintained. ~~the control bank insertion limits restrict the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.~~

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Operation within the AFD, QPTR, shutdown and control bank insertion and alignment LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown bank and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown bank shall be at or above the insertion limit and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and the shutdown bank (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The control bank insertion limits also limit the reactivity worth of an ejected control bank rod.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. Reactor Coolant System pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the control bank insertion limits affect safety analysis involving core reactivity and power distributions (Refs. 4, 5, 6, and 7).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Refs. 4, 5, 6, and 7).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits, together with AFD, QPTR and the control and shutdown bank alignment limits, ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Refs. 4, 5, 6, and 7).

The control bank insertion, sequence and overlap limits satisfy Criterion 2 of the NRC Policy Statement, in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is limited, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

The rod insertion limit monitor is used to verify control rod insertion on a continuous basis and will provide an alarm whenever the control bank insertion deviates from the rod insertion limits specified in the COLR. Verification that the control banks are within the insertion limit is made every 12 hours (SR 3.1.6.2). When the rod insertion limit monitor is inoperable a verification that the rod positions are within the limit must be made more frequently (SR 3.1.6.3).

The LCO is modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.3. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

(continued)

BASES

APPLICABILITY

The control bank insertion, sequence, and overlap limits shall be maintained with the reactor in MODE 1 and MODE 2 with $k_{\text{eff}} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODE 2 with $k_{\text{eff}} < 1.0$ and MODES 3, 4, 5, and 6 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

ACTIONS

A.1.1, A.1.2, and A.2

When the control banks are outside the acceptable insertion limits, out of sequence, or in the wrong overlap configuration, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position; or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM within 1 hour is required, since the SDM in MODES 1 and 2 is normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). ~~"SHUTDOWN MARGIN (SDM)"~~ has been upset. If control banks are not within their limits, then SDM will be verified by performing a reactivity balance calculation, taking into account RCS boron concentration, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

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(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability. Thus, the allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems.

B.1

If Required Actions A.1 and A.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with $K_{eff} < 1.0$, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. The Frequency of within 4 hours prior to achieving criticality ensures that the estimated control bank position is within the limits specified in the COLR shortly before criticality is reached.

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SR 3.1.6.2

With an OPERABLE bank insertion limit monitor (i.e., the PPCS),

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SR 3.1.6.2

(continued)

BASES

(169)

~~With an OPERABLE bank insertion limit monitor,~~ verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.6.3

When the insertion limit monitor (i.e., the PPCS) becomes inoperable, no control room alarm is available between the normal 12 hour frequency to alert the operators of a control bank not within the insertion limits. A reduction of the Frequency to every 4 hours provides sufficient monitoring of control rod insertion when the monitor is inoperable. Verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

This SR is modified by a Note that states that performance of this SR is only necessary when the rod insertion limit monitor is inoperable.

SR 3.1.6.4

When control banks are maintained within their insertion limits as required by SR 3.1.6.2 and SR 3.1.6.3 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, 28, 29, and 32, Issued for comment July 10, 1967.
 2. 10 CFR 50.46.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 15.1.5.
 5. UFSAR, Section 15.4.1.
 6. UFSAR, Section 15.4.2.
 7. UFSAR, Section 15.4.6.
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B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

The OPERABILITY (i.e., trippability), including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SHUTDOWN MARGIN (SDM). Rod position indication is required to assess OPERABILITY and misalignment.

According to the Atomic Industrial Forum (AIF) GDC 12 and 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are movable neutron absorbing devices which are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

(continued)

BASES

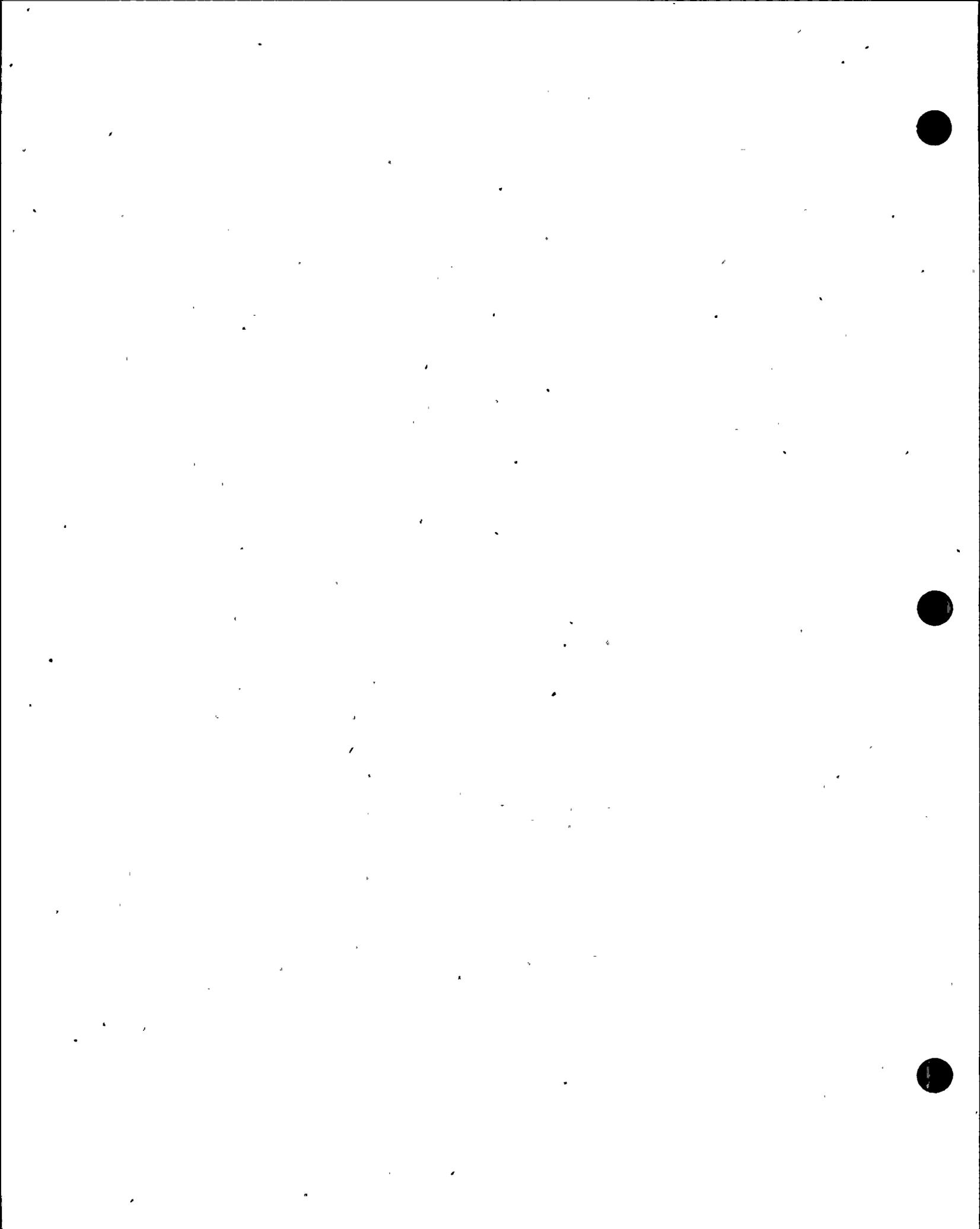
BACKGROUND
(continued)

The RCCAs are divided among control banks and a shutdown bank. Control banks are used to compensate for changes in reactivity due to variations in operating conditions of the reactor such as coolant temperature, power level, boron or xenon concentration. The shutdown bank provides additional shutdown reactivity such that the total shutdown worth of the bank is adequate to provide shutdown for all operating and hot zero power conditions with the single RCCA of highest reactivity worth fully withdrawn. Each bank is further subdivided into groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion but always within one step of each other. There are four control banks and one shutdown bank at Ginna Station.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Microprocessor Rod Position Indication (MRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{1}{8}$ inch), but if a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

(continued)



BASES

BACKGROUND
(continued)

The MRPI System also provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. The MRPI system consists of one digital detector assembly per rod. All the detector assemblies consist of one coil stack which is multiplexed and becomes input to two redundant MRPI signal processors. Each signal processor independently monitors all rods and senses a rod bottom for any rod. The MRPI system directly senses rod position in intervals of 12 steps for each rod. The digital detector assemblies consist of 20 discrete coil pairs spaced at 12-step intervals. The true rod position is always within ± 8 steps of the indicated position (± 6 steps due to the 12-step interval and ± 2 steps transition uncertainty due to processing and coil sensitivity). With an indicated deviation of 12 steps between the group step counter and MRPI, the maximum deviation between actual rod position and the demand position would be 20 steps, or 12.5 inches.

APPLICABLE
SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth limits, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limit," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

(continued)

BASES

LCO

LCO 3.1.7 specifies that the MRPI System and the Bank Demand Position Indication System be OPERABLE. For the control rod position indicators to be OPERABLE requires the following:

- a. For the MRPI System there are no failed coils and rod position indication is available on the MRPI screen (in either the control room or relay room) or the plant process computer system; and
- b. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the MRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the MRPI System as required by SR 3.1.7.1 indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of control rod bank position. A deviation of less than the allowable 12 step agreement limit, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis.

The MRPI system is designed with error detection such that when a fault occurs in the binary data received from the coil stacks or processing unit an alarm is annunciated at the MRPI display. When the fault clears, the system provides self validation of data integrity and returns to its normal display mode. Because of the digital nature of the system and its inherent diagnostic features, intermittent data alarms can mask position indication and generate the perception that a single rod position is unmonitored. For a single rod position indication failure, MRPI is considered OPERABLE if a fault occurs and clears within five minutes and the indicated position is within expected values.

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

(continued)

BASES

APPLICABILITY The requirements on the MRPI and step counters are only applicable in MODE 1 and MODE 2 with $K_{\text{off}} \geq 1.0$ (consistent with LCO 3.1.4 and LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which the reactor is critical, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODE 2 with $K_{\text{off}} < 1.0$ and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

~~**ACTIONS** The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable MRPI per group and each demand position indicator per~~

~~bank. **ACTIONS** The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each group with no more than one inoperable rod position indicator in the group and for each bank with no more than one inoperable demand position indicator in the bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.~~

~~With more than one inoperable rod position indicator per group or more than one inoperable demand position indicator per bank, the plant must enter LCO 3.0.3.~~

A.1

When one MRPI per group fails, the position of the rod can still be determined by use of the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

(continued)

BASES

(continued)

BASES

ACTIONS
(continued)

A.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

~~When one or more rods with inoperable position indicators (i.e., MRPI) have been moved > 24 steps in one direction since the position was last determined:~~

~~These Required Actions ensure that when one or more rods with inoperable position indicators (i.e., MRPI) have been moved > 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.~~

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The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

Acceptable verification of rod position within 4 hours re-initiates the clock for Required Action A.1.

~~THE FOLLOWING TEXT WAS MOVED~~

~~If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP, if one or more rods are misaligned by more than 24 steps.~~

~~THE PRECEDING TEXT WAS MOVED~~

(continued)

BASES

ACTIONS
(continued)

C.1.1 and C.1.2

60

With one demand position indicator per bank inoperable, the rod positions can be determined by the MRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps from the OPERABLE demand position indicator for that bank within the allowed Completion Time of once every 8 hours is adequate. This ensures that most withdrawn and least withdrawn rod are no more than 24 steps apart which is less than the accident analysis assumption of 25 steps. This verification can be an examination of logs, administrative controls, or other information that shows that all MRPIs in the affected bank are OPERABLE.

C.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position will not cause core peaking to approach the core peaking factor limits.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 2 with $K_{eff} < 1.0$ within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

ACTIONS E-1
(continued)

E-1

(149)

With more than one MRPI per group inoperable for one or more groups or more than one demand position indicator per bank inoperable for one or more banks, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1

Verification that the MRPI agrees with the group demand position within 12 steps for the full indicated range of rod travel ensures that the MRPI is operating correctly. Since the MRPI does not display the actual shutdown rod positions between 0 and 230 steps, only points within the indicated ranges are required in comparison.

This Surveillance is performed during a plant outage or during plant startup, prior to reactor criticality after each removal of the reactor head due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 12 and 13, Issued for comment July 10, 1967.
 2. UFSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in the design and analysis;
- c. Verify the assumptions used to predict plant response;
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power, power ascension, and at power operation; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

(continued)

BASES (continued)

BACKGROUND
(continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS performed at Ginna Station for reload fuel cycles in MODE 2 include:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and
- d. Isothermal Temperature Coefficient (ITC).

These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance as described below.

- a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, bank D is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff} = 1.0$), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test could violate LCO 3.1.3, "Moderator Temperature Coefficient (MTC)."

(continued)

BASES (continued)

BACKGROUND
(continued)

- (16)
- b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ fully inserted into the core. This test is used to measure the differential boron worth. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. ~~The reactivity resulting from each incremental bank movement is measured with a reactivity computer (i.e., the PPCS).~~ The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The differential boron worth is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits;" LCO 3.1.5, "Shutdown Bank Insertion Limit;" or LCO 3.1.6, "Control Bank Insertion Limits."
- (17)
- c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has two alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. ~~The reactivity changes are measured with a reactivity computer (i.e., the PPCS).~~ This sequence is repeated for the remaining control banks. The second method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

(continued)

BASES (continued)

BACKGROUND
(continued)

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- d. The ITC Test measures the ITC of the reactor. This test is performed at HZP using the Slope Method. The Slope Method varies RCS temperature in a slow and continuous manner. ~~The reactivity change is measured with a reactivity computer as a function of the temperature change. The reactivity change is measured with a reactivity computer (i.e., the PPCS) as a function of the temperature change.~~ The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The Moderator Temperature Coefficient (MTC) at BOL, 70% RTP and at EOL is determined from the measured ITC. This test satisfies the requirements of SR 3.1.3.1 and SR 3.1.3.2. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."
-

APPLICABLE
SAFETY ANALYSES

The fuel is protected by multiple LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of these LCOs, that are excepted by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 3). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The UFSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Reference 4 summarizes the initial zero, low power, and power tests. Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines and established industry practices which are consistent with the PHYSICS TESTS described in References 5 and 6. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. The requirements specified in the following LCOs may be suspended for PHYSICS TESTING:

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limit";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality".

When these LCOs are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to $\leq 5\%$ RTP, the reactor coolant temperature is kept $\geq 530^\circ\text{F}$, and SDM is within the limits specified in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the plant safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits to conduct PHYSICS TESTS in MODE 2, to verify certain core physics parameters. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

- a. THERMAL POWER is maintained $\leq 5\%$ RTP;
 - b. RCS lowest loop average temperature is $\geq 530^\circ\text{F}$; and
 - c. SDM is within the limits specified in the COLR.
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(continued)

BASES (continued)

~~APPLICABILITY~~ This LCO is applicable when performing low power PHYSICS TESTS. ~~APPLICABILITY~~ This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

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ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification within 1 hour.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits since a MODE change has occurred. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS loop with the lowest T_{avg} is < 530°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 530°F could violate the assumptions for accidents analyzed in the safety analyses.

(continued)

BASES (continued)

(continued)

BASES (continued)

ACTIONS
(continued)

D.1

If Required Action C.1 cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from MODE 2 in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 7 days prior to criticality. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 7 day time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is $\geq 530^{\circ}F$ will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

(continued)

SR 3.1.8.3

Verification that THERMAL POWER is < 5% RTP using the NIS detectors will ensure that the plant is not operating in a condition that could invalidate the safety analyses. -

(continued)

~~Verification that the THERMAL POWER < 5% RTP using the NIS detectors will ensure that the plant is not operating in a condition that could invalidate the safety analyses.~~ Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

The SDM is verified by comparing the RCS boron concentration to a SHUTDOWN MARGIN requirement curve that was generated by taking into account estimated RCS boron concentrations, core power defect, control bank position, RCS average temperature, fuel burnup based on gross thermal energy generation, xenon concentration, samarium concentration, and isothermal temperature coefficient (ITC).

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
 4. UFSAR, Section 14.6.
 5. Letter from R. W. Kober (RGE) to T. E. Murley (NRC), Subject: "Startup Reports," dated July 9, 1984.
 6. Letter from J. P. Durr (NRC) to B. A. Snow (RGE), Subject: "Inspection Report No. 50-244/88-06," dated April 28, 1988.
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3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor (F_Q(Z))

LCO 3.2.1 F_Q(Z) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F _Q (Z) not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% F _Q (Z) exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce AFD acceptable operation limits ≥ 1% for each 1% F _Q (Z) exceeds limit.	8 hours
	<u>AND</u>	
	A.3 Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% F _Q (Z) exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Reduce Overpower ΔT and Overtemperature ΔT trip setpoints ≥ 1% for each 1% F _Q (Z) exceeds limit.	72 hours
	<u>AND</u>	(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 Perform SR 3.2.1.1 or SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify measured values of F _a (Z) are within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

(continued)

ACTIONS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.2</p> <p>-----NOTE----- Only required to be performed if one power range channel is inoperable with THERMAL POWER \geq 75% RTP. -----</p> <p>Verify measured values of F_a(Z) are within limits specified in the COLR.</p>	<p>Once within 24 hours and every 24 hours thereafter</p>



3.2 POWER DISTRIBUTION LIMITS

 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

 LCO 3.2.2 $F_{\Delta H}^N$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_{\Delta H}^N$ not within limit.	A.1 Reduce THERMAL POWER $\geq 1\%$ RTP for each 1% $F_{\Delta H}^N$ exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux-High trip setpoints $\geq 1\%$ for each 1% $F_{\Delta H}^N$ exceeds limit.	72 hours
	<u>AND</u>	
	A.3 Reduce Overpower ΔT and Overtemperature ΔT trip setpoints $\geq 1\%$ for each 1% $F_{\Delta H}^N$ exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Perform SR 3.2.2.1 or SR 3.2.2.2.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify $F_{\Delta H}^N$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter
SR 3.2.2.2 -----NOTE----- Only required to be performed if one power range channel is inoperable with THERMAL POWER \geq 75% RTP. ----- Verify $F_{\Delta H}^N$ is within limits specified in the COLR.	Once within 24 hours and every 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD monitor alarm shall be OPERABLE and AFD:

- a. Shall be maintained within the target band about the target flux difference with THERMAL POWER \geq 90% RTP. The target band is specified in the COLR.
- b. May deviate outside the target band with THERMAL POWER $<$ 90% RTP but \geq 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is \leq 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.
- c. May deviate outside the target band with THERMAL POWER $<$ 50% RTP.

-----NOTES-----

1. The AFD shall be considered outside the target band when the average of four OPERABLE excore channels indicate AFD to be outside the target band. If one excore detector is out of service, the remaining three detectors shall be used to derive the average.
2. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with THERMAL POWER \geq 50% RTP, and AFD outside the target band.
3. Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with THERMAL POWER $>$ 15% RTP and $<$ 50% RTP, and AFD outside the target band.
4. A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6.

APPLICABILITY: MODE 1 with THERMAL POWER $>$ 15% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. THERMAL POWER \geq 90% RTP.</p> <p><u>AND</u></p> <p>AFD not within the target band.</p>	<p>A.1 Restore AFD to within target band.</p>	<p>15 minutes</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Initiate action to reduce Reduce THERMAL POWER to < 90% RTP.</p>	<p>Immediately 15 minutes</p>
<p>C. THERMAL POWER < 90% RTP and \geq 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours.</p> <p><u>OR</u></p> <p>THERMAL POWER < 90% RTP and \geq 50% RTP with AFD not within the target band and not within the acceptable operation limits.</p>	<p>C.1 Initiate action to reduce Reduce THERMAL POWER to < 50% RTP.</p>	<p>Immediately 30 minutes</p>

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. THERMAL POWER $\geq 90\%$ RTP. <u>AND</u> AFD monitor alarm inoperable.	D.1 Perform SR 3.2.3.1.	Once every 15 minutes
E. THERMAL POWER < 90% RTP. <u>AND</u> AFD monitor alarm inoperable.	E.1 Perform SR 3.2.3.2.	Once every 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.3.1 NOTES</p> <p>1. Verify AFD monitor is OPERABLE.</p>	<p>12 hours</p>
<p>SR 3.2.3.2 NOTES</p> <p>1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER \geq 90% RTP.</p> <p>2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available.</p> <p>-----</p> <p>Verify AFD is within limits and log AFD for each OPERABLE excore channel.</p>	<p>Once within 15 minutes and every 15 minutes thereafter</p>
<p>SR 3.2.3.23 NOTES</p> <p>1. Only required to be performed if AFD monitor alarm is inoperable when THERMAL POWER < 90% RTP.</p> <p>2. Assume logged values of AFD exist during the preceding 24 hour time interval if actual AFD values are not available.</p> <p>-----</p> <p>Verify AFD is within limits and log AFD for each OPERABLE excore channel.</p>	<p>Once within 1 hour and every 1 hour thereafter</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>(63) SR 3.2.3.33 2.3.4 Update target flux difference.</p>	<p>Once within 31 EFPD after each refueling</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>
<p>(63) SR 3.2.3.43 2.3.5 -----NOTE----- ----- The initial target flux difference after each refueling may be determined from design predictions. ----- Determine, by measurement, the target flux difference.</p>	<p>Once within 31 EFPD after each refueling</p> <p><u>AND</u></p> <p>92 EFPD thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR monitor alarm shall be OPERABLE and QPTR shall be ≤ 1.02 .

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Limit THERMAL POWER to $\geq 3\%$ below RTP for each 1% of QPTR > 1.00.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.4.1 and limit THERMAL POWER to $\geq 3\%$ below RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>	
	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Actions A.1 and A.2
	<u>AND</u>	
		Once per 7 days thereafter
	<u>AND</u>	(continued)

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A.4 Reactivate safety analyses and confirm results remain valid for the duration of operation under this condition.

Prior to increasing THERMAL POWER above the limit of Required Action A.1

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p> <p>(62)</p> <p>(63)</p>	<p>A.2⁵</p> <p>-----NOTE----- Perform Required Action A.2⁵ only after Required Action A.3⁴ has verified that the hot channel factors are within limits. -----</p> <p>Calibrate/Normalize excore detector instrumentation to eliminate the indicated tilt.</p> <p>AND</p>	<p>is complete.</p> <p>Prior to increasing THERMAL POWER above the limit of Required Actions A.1 and A.2</p> <p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p><i>A₂</i></p> <p>-----NOTES-----</p> <p>1. Only required to be performed if the cause of the QPTR alarm is not associated with instrumentation alignment.</p> <p>2. Perform Required Action A₂ only after must be completed when Required Action A₂ is completed implemented.</p> <p>3. Only one of the Completion Times, whichever becomes applicable first, must be met.</p> <p>-----</p> <p>Perform SR 3.2.1.1 and SR 3.2.2.1.</p>	<p>impossible if R</p> <p>completed and Note 1, also see act 4.1.2</p> <p>Within 24 hours after reaching RTP</p> <p>OR</p> <p>Within 48 hours after increasing achieving equilibrium conditions with THERMAL POWER increased above the limits of Required Actions A.1 and A.2</p>

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to \leq 50% RTP.	4 hours

(continued)

C. QPTR monitor alarm inoperable.	C.1 Perform SR 3.2.4.2	Once within 24 hours and every 24 hours thereafter
	<u>OR</u>	
	C.2 Perform SR 3.2.1.2 and SR 3.2.2.2	Once within 24 hours and every 24 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.4.1 <u>NOTES</u> Verify QPTR monitor alarm is OPERABLE.	12 hours

ACTIONS (continued)

SURVEILLANCE	FREQUENCY
<p>THE FOLLOWING TEXT WAS MOVED</p> <p>SR 3.2.4.2 -----NOTES-----</p> <p>1. THE PRECEDING TEXT WAS MOVED With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR.</p> <p>2. With one power range channel inoperable and THERMAL POWER \geq 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2.</p> <p>-----</p> <p>Verify QPTR is within limit by calculation.</p>	<p>7 days</p>

(continued)

<p>SR 3.2.4.3 -----NOTES-----</p> <p>1. Only required to be performed if the QPTR monitor alarm is inoperable.</p> <p>2. With one power range channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR.</p> <p>3. With one power range channel inoperable and THERMAL POWER \geq 75% RTP, perform SR 3.2.1.2 and SR 3.2.2.2.</p> <p>-----</p> <p>Verify QPTR is within limit by calculation.</p>	<p>Once within 24 hours and every 24 hours thereafter</p>
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor (F₀(Z))

BASES

BACKGROUND

The purpose of the limits on the values of F₀(Z) is to limit the local (i.e., pellet) peak power density. The value of F₀(Z) varies along the axial height of the core (Z).

F₀(Z) is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions adjusted for uncertainty. Therefore, F₀(Z) is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis.

F₀(Z) is sensitive to fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

F₀(Z) is measured periodically using the incore detector system. Measurements are generally taken with the core at or near steady state conditions. With the measured three dimensional power distributions, it is possible to determine a measured value for F₀(Z). However, because this value represents a steady state condition, it does not include variations in the value of F₀(Z), which are present during a nonequilibrium situation such as load following when the plant changes power level to match grid demand peaks and valleys.

Core monitoring and control under transient conditions (i.e., Condition 1 events as described in Reference 1) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion, Sequence and Overlap Limits.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

Limits on F_Q(Z) preclude core power distributions that violate the following fuel design criteria:

- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 2);
- c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 3); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SHUTDOWN MARGIN (SDM) with the highest worth control rod stuck fully withdrawn (Ref. 4).

Limits on F_Q(Z) ensure that the value of the total peaking factor assumed as an initial condition in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

The F_Q(Z) limits provided in the COLR are based on the limits used in the LOCA analysis. F_Q(Z) limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the F_Q(Z) assumed in safety analyses for other accidents because of the requirements set forth in 10 CFR 50.46 (Ref. 2) and ECCS model development in accordance with the required features of the ECCS evaluation models provided in 20 CFR 50, Appendix K (Ref. 5). Therefore, this LCO provides conservative limits for other accidents.

F_Q(Z) satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO

The F_Q(Z) shall be maintained within the limits of the relationships provided in the COLR.

The F_Q(Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA (Refs. 6 and 7).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F_Q(Z) limits. If F_Q(Z) cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for F_Q(Z) may produce unacceptable consequences if a design basis event occurs while F_Q(Z) is outside its specified limits.

APPLICABILITY

The F_Q(Z) limits must be maintained while in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ for each 1% by which F_Q(Z) exceeds its limit maintains an acceptable absolute power density. ~~The 15-minute Completion Time begins at the time the analysis of an incore flux map verifies the limit is exceeded and the shift supervisor has been notified.~~ The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

ACTIONS
(continued)

A.2

When core peaking factors are sufficiently high that LCO 3.2.1 does not permit operation at RTP, the acceptable operation limits for AFD are reduced. The acceptable operation limits are reduced 1% for each 1% by which F_Q(Z) exceeds its limit. For example, if the measured F_Q(Z) exceeds the limit by 3% and the acceptable operation limits for AFD are ± 11% at 90% RTP and ± 31% at 50% RTP, then the revised AFD Acceptable Operation Limits would be ± 8% at 90% RTP and ± 28% at 50% RTP. This ensures a near constant maximum linear heat rate in units of kilowatts per foot at the acceptable operation limits. The Completion Time of 8 hours for the change in setpoints is sufficient, considering the small likelihood of a severe transient in this relatively short time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

A reduction of the Power Range Neutron Flux-High trip setpoints by ≥ 1% for each 1% by which F_Q(Z) exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since this trip setpoint helps protect reactor core safety limits. This reduction shall be made as follows, given an F_Q(Z) limit of 2.32, a measured F_Q(Z) of 2.4, and a Power Range Neutron Flux-High setpoint of 108%, the Power Range Neutron Flux-High setpoint must be reduced by at least 3.4% to 104.6%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

ACTIONS
(continued)

A.4

Reduction in the Overpower ΔT and Overttemperature ΔT trip setpoints by $\geq 1\%$ for each 1% by which F_a(Z) exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since these trip setpoints help protect reactor core safety limits. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.5

Verification that F_a(Z) has been restored to within its limit by performing SR 3.2.1.1 or SR 3.2.1.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.5 cannot be met within their associated Completion Times, the plant must be placed in a MODE or Condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1

Verification that F_Q(Z) is within its limit involves increasing the measured values of F_Q(Z) to allow for manufacturing tolerance and measurement uncertainties and then making a comparison with the limits. These limits are provided in the COLR. Specifically, the measured value of the Heat Flux Hot Channel Factor (F_Q^M) is increased by 3% to account for fuel manufacturing tolerances and by 5% for flux map measurement uncertainty for a full core flux map using the movable incore detector flux mapping system. This procedure is equivalent to increasing the directly measured values of F_Q(Z) by 1.0815% before comparing with LCO limits.

Performing the Surveillance in MODE 1 prior to THERMAL POWER exceeding 75% RTP after each refueling ensures that F_Q(Z) is within limit when RTP is achieved and provides confirmation of the nuclear design and the fuel loading pattern.

The Frequency of 31 EFPD is adequate for monitoring the change of power distribution with core burnup because the power distribution changes relatively slowly for this amount of fuel burnup. Accordingly, this Frequency is short enough that the F_Q(Z) limit cannot be exceeded for any significant period of duration. ~~When the plant is already performing SR 3.2.1.2 to satisfy other requirements, SR 3.2.1.2 does not need to be suspended in order to perform SR 3.2.1.1 since the performance of SR 3.2.1.2 meets the requirements of SR 3.2.1.1.~~

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SR 3.2.1.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.1.2 at a frequency of 24 hours provides an accurate alternative means for ensuring that F_0 remains within limits and the core power distribution is consistent with the safety analyses. A frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is $\geq 75\%$ RTP.

REFERENCES

1. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 2. 10 CFR 50.46.
 3. UFSAR, Section 15.4.5.1.
 4. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.
 5. 10 CFR 50, Appendix K.
 6. UFSAR, Section 15.6.4.1.
 7. UFSAR, Section 15.6.4.2.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location in the core during either normal operation or a postulated accident analyzed in the safety analyses.

$F_{\Delta H}^N$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F_{\Delta H}^N$ is a measure of the maximum total power produced in a fuel rod. The $F_{\Delta H}^N$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for departure from nucleate boiling (DNB).

$F_{\Delta H}^N$ is sensitive to fuel loading patterns, control bank insertion, and fuel burnup. $F_{\Delta H}^N$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F_{\Delta H}^N$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F_{\Delta H}^N$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. Therefore, these LCOs preserve core limits on a continuous basis. $F_{\Delta H}^N$ and the QPTR LCO limit the radial component of the peaking factors.

(continued)

BASES

BACKGROUND
(continued)

The COLR provides peaking factor limits that ensure that the design basis value for departure from nucleate boiling ratio (DNBR) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio. All DNB limited transient events are assumed to begin with an $F_{\Delta H}^N$ value that satisfies the LCO requirements.

The design method employed to meet the DNB design criterion for fuel assemblies is the Improved Thermal Design Procedure (ITDP). With the ITDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, ITDP design limit DNBR values are determined in order to meet the DNB design criterion.

The ITDP design limit DNBR values are 1.34 and 1.33 for the typical and thimble cells, respectively, for fuel analyses with the WRB-2 correlation.

DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR values are 1.52 and 1.51 for the typical and thimble cells, respectively.

(continued)

BASES

 BACKGROUND
 (continued)

For both the WRB-1 and WRB-2 correlations, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation applies in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

 APPLICABLE
 SAFETY ANALYSES

Limits on $F_{\Delta H}^N$ preclude core power distributions that exceed the following fuel design limits:

- (169)
- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;
 - b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);
 - c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and
 - d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

 (continued)

BASES

 APPLICABLE
 SAFETY ANALYSES
 (continued)

For transients that may be DNB limited, the Reactor Coolant System flow and $F_{\Delta H}^N$ are the core parameters of most importance. The limits on $F_{\Delta H}^N$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency (i.e., Condition 1 events as described in Reference 4). The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion.

The allowable $F_{\Delta H}^N$ limit increases with decreasing power level. This functionality in $F_{\Delta H}^N$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F_{\Delta H}^N$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F_{\Delta H}^N$ as a function of power level defined by the COLR limit equation.

The LOCA safety analysis indirectly models $F_{\Delta H}^N$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 1).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F_{\Delta H}^N$ is measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion, Sequence and Overlap Limits.

$F_{\Delta H}^N$ satisfies Criterion 2 of the NRC Policy Statement.

 (continued)

BASES

LCO

$F_{\Delta H}^N$ shall be maintained within the limits of the relationship provided in the COLR.

The $F_{\Delta H}^N$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for DNB.

The limiting value of $F_{\Delta H}^N$, described by the equation contained in the COLR, is the design radial peaking factor used in the plant safety analyses.

A power multiplication factor in this equation includes an additional margin for higher radial peaking from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F_{\Delta H}^N$ is allowed to increase 0.3% for every 1% RTP reduction in THERMAL POWER.

APPLICABILITY

The $F_{\Delta H}^N$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to $F_{\Delta H}^N$ in MODES 2, 3, 4, and 5 have significant margin to DNB, and therefore, there is no need to restrict $F_{\Delta H}^N$ in these modes.

(continued)



BASES

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its limit maintains an acceptable DNBR margin. When the $F_{\Delta H}^N$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F_{\Delta H}^N$ value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNBR limiting event occurs. Reducing THERMAL POWER increases the DNBR margin and does not likely cause the DNBR limit to be violated in steady state operation. ~~The 15-minute Completion Time begins at the time the analysis of an incore flux map verifies the limit is exceeded and the shift supervisor has been notified.~~ The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2

A reduction of the Power Range Neutron Flux-High ~~Neutron Flux-High~~ trip setpoints by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its specified limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions and ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. This reduction shall be made as follows, given that the $F_{\Delta H}^N$ limit is exceeded by 3% and the Power Range Neutron Flux-High setpoint is 108%, the Power Range Neutron Flux-High setpoint must be reduced by at least 3% to 105%. The Completion Time of 72 hours is sufficient, considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with required action A.1.

 (continued)

BASES

ACTIONS
(continued)

A.3

Reduction in the Overpower ΔT and Overtemperature ΔT trip setpoints by $\geq 1\%$ for each 1% by which $F_{\Delta H}^N$ exceeds its limit, ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F_{\Delta H}^N$ has been restored within its limit by performing SR 3.2.2.1 or SR 3.2.2.2 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1 ensures that the cause that led to the $F_{\Delta H}^N$ exceeding its limit is corrected, and core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If the Required Actions of A.1 through A.4 cannot be met within their associated Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

The allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.2.1

The value of $F_{\Delta H}^N$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F_{\Delta H}^N$ from the measured flux distributions. The measured value of $F_{\Delta H}^N$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F_{\Delta H}^N$ limit.

After each refueling, $F_{\Delta H}^N$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F_{\Delta H}^N$ limits are met at the beginning of each fuel cycle.

The Frequency of 31 EFPD is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}^N$ limit cannot be exceeded for any significant period of operation. When the plant is already performing SR 3.2.2.2 to satisfy other requirements, SR 3.2.2.1 does not need to be suspended in order to perform SR 3.2.2.1 since the performance of SR 3.2.2.2 meets the requirements of SR 3.2.2.1.

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SR 3.2.2.2

During power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables.

With an NIS power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.2.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that $F_{\Delta H}^N$ remains within limits and the core power distribution is consistent with the safety analyses. A Frequency of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map.

(continued)

BASES

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REQUIREMENTS

SURVEILLANCE SR 3.2.2.2 (continued)

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is \geq 75% RTP.



REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.5.1.
3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10 1967.
4. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.



(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during plant maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) and QUADRANT POWER TILT RATIO (QPTR) LCOs limit the radial component of the peaking factors.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Ref. 1) entails:

- a. Establishing an envelope of allowed power shapes and power densities;
- b. Devising an operating strategy for the cycle that maximizes plant flexibility (maneuvering) and minimizes axial power shape changes;
- c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and
- d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events (Ref. 2). This ensures that fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 event is the loss of coolant accident. The most significant Condition 3 event is the loss of flow accident. The most significant Condition 2 events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 accidents, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator to help define the power profile from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom excore neutron detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\% \Delta$ flux or $\% \Delta I$.

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With THERMAL POWER \geq 90% RTP (i.e., Part A of this LCO), the AFD must be kept within the target band about the target flux difference. ~~The target band is provided in the COLR~~
With the AFD outside the target band with THERMAL POWER \geq 90% RTP, the assumptions of the accident analyses may be violated. With THERMAL POWER $<$ 90% RTP, the AFD may be outside the target band provided that the deviation time is restricted.

It is intended that the plant is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is \geq 50% RTP and $<$ 90% RTP (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours when $>$ 15% RTP, is allowed during which the plant may be operated outside of the target band but within the acceptable operation limits provided in the COLR. The cumulative penalty time is the sum of penalty times as calculated by Notes 2 and 3 of this LCO.

(continued)

BASES

LCO
(continued)

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The reduced penalty deviation time accumulation rate reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the Plant Process Computer System (PPCS) is nominally once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time. The inoperability of this monitor requires independent verification that AFD remains within limit and that the peaking factors assumed in the accident analyses remain valid.

(continued)

BASES

LCO
(continued)

This LCO is modified by four Notes. The first Note states the conditions necessary for declaring the AFD outside of the target band. The required target band varies with axial burnup distribution, which in turn varies with the core average accumulated burnup. The target band defined in the COLR may provide one target band for the entire cycle or more than one band, each to be followed for a specific range of cycle burnup. The average of the four OPERABLE excore detectors is used to determine when AFD is outside the target band. If one excore detector is out of service, the remaining three detectors are used to derive the average AFD. The second and third Notes describe how the cumulative penalty deviation time is calculated. The second Note states that with THERMAL POWER \geq 50% RTP the penalty deviation time is accumulated at the rate of 1 minute for each 1 minute of power operation with AFD outside the target band. The third Note states that with THERMAL POWER $>$ 15% RTP and $<$ 50% RTP the penalty deviation time is accumulated at the rate of 0.5 minutes for each 1 minute of power operation with AFD outside the target band. The cumulative penalty time is the sum of penalty times from Notes 2 and 3 of this LCO. The fourth Note addresses AFD outside of the target band during surveillances. For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

(continued)



BASES

APPLICABILITY AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 1). Above 15% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. Also, low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. The value of the AFD in these conditions does not affect the consequences of the design basis events.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER \geq 90% RTP, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If Required Action A.1 is not completed with the required Completion Time of 15 minutes, the axial xenon distribution starts to become skewed. ~~Immediately initiating action to reduce~~ Reducing THERMAL POWER to $<$ 90% RTP places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

(45)

~~Immediately initiating the reduction in~~ The allowed Completion Time of 15 minutes to reduce THERMAL POWER to $<$ 90% RTP allows for a controlled reduction in power without allowing the plant to remain in an unanalyzed condition for an extended period of time.

(continued)



BASES

(continued)



BASES

ACTIONS
(continued)C.1

This Required Action must be implemented with THERMAL POWER < 90% RTP but \geq 50% RTP if either the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

With THERMAL POWER < 90% RTP but \geq 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of xenon buildup and decay. ~~Immediately initiating the reduction~~ Reducing THERMAL POWER to a power level < 50% RTP will put the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter.

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If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits. ~~The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.~~

D.1

When the AFD monitor alarm is inoperable and THERMAL POWER is \geq 90% RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every 15 minutes to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 15 minutes is adequate to ensure that the AFD is within its limits at high THERMAL POWER levels and is consistent with the Completion Time for restoring AFD to within limits (Condition A).

(continued)

BASES

ACTIONS
(continued)E.1

When the AFD monitor alarm is inoperable and THERMAL POWER is < 90% RTP, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time at a frequency of every hour to ensure that the plant does not operate in an unanalyzed condition. A Completion Time of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SURVEILLANCE
REQUIREMENTSSR 3.2.3.1

63 The AFD This SR is monitored on a continuous basis using the Plant Process Computer System (PPCS) the verification that has an the AFD monitor alarm is OPERABLE. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control annunciator immediately if the average AFD This is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals normally accomplished by introducing a signal into the plant process computer to verify control room annunciation of AFD not within the previous 24 hour target band. The computer, also sends an alarm message when the cumulative penalty deviation time is \geq 1 hour within the previous 24 hours is sufficient to ensure OPERABILITY of the AFD monitor since under normal plant operation, the AFD is not expected to significantly change. —

This alarm message does not clear until the cumulative penalty deviation time SR 3.2.3.2

The AFD is < 1 hour within the previous 24 hours monitored on a continuous basis using the Plant Process Computer System (PPCS) that has an AFD monitor alarm.

With the AFD monitor alarm inoperable, the AFD measurement determined by the The PPCS must be independently monitored to detect operation outside determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm

(continued)

BASES

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message and a main control board annunciator immediately if the average AFD is outside the target band and to compute the then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.

(continued)

BASES

SURVEILLANCE SR 3.2.3.2 (continued)
REQUIREMENTS

(63)

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at $\geq 90\%$ RTP, the AFD measurement is monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. The AFD should be monitored and logged more frequently during periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SURVEILLANCE SR 3.2.3.1 (continued)
REQUIREMENTS

~~SR 3.2.3.1 is modified by two Notes. SR 3.2.3.2 is modified by two Notes.~~ The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER $\geq 90\%$ RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

SR 3.2.3.23.2.3.3

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The AFD is monitored on a continuous basis using the PPCS that has an AFD monitor alarm. The PPCS determines the 1 minute average of the OPERABLE excore detector outputs and provides an alarm message and a main control board annunciator immediately if the average AFD is outside the target band and then re-alarms when the cumulative penalty deviation time reaches 15 minute intervals within the previous 24 hours. ~~The computer, also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.~~

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.2.3.2~~ (continued)
~~REQUIREMENTS~~

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~~With the AFD monitor alarm inoperable, the AFD measurement determined by the PPGS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. The computer also sends an alarm message when the cumulative penalty deviation time is ≥ 1 hour within the previous 24 hours. This alarm message does not clear until the cumulative penalty deviation time is < 1 hour within the previous 24 hours.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS SR 3.2.3.3 (continued)

(62)

With the AFD monitor alarm inoperable, the AFD measurement determined by the PPCS must be independently monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at < 90% RTP, but > 15% RTP, the AFD measurement is monitored at a Surveillance Frequency of 1 hour to ensure that the AFD is within its limits. The Frequency of 1 hour is adequate since the AFD may deviate from the target band for up to 1 hour using the methodology of Notes 2 and 3 of this LCO to calculate the cumulative penalty deviation time before corrective action is required. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

(63)

SR 3.2.3.3 is modified by two Notes. The first Note states that this surveillance is only required to be performed when the AFD monitor alarm is inoperable with THERMAL POWER < 90% RTP. The second Note states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time if AFD values cannot be obtained from the PPCS. Inoperability of the alarm does not necessarily prevent the actual AFD values from being available (e.g., from the computer logs or hand logs). AFD values for the preceding 24 hours can be obtained from the hourly PPCS printouts or hand logs.

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SR 3.2.3.3.2.3.4

This Surveillance requires that the target flux difference be updated at a Frequency of 31 effective full power days (EFPD) to account for small changes that may occur in the target flux differences in that period due to burnup.

(continued)

D
BASES

SURVEILLANCE
REQUIREMENTSSR 3-2-3-33 2-3-4 (continued)

63 There are two methods by which this update can be completed. The first method requires measuring the target flux difference in accordance with SR 3-2-3-43 2-3-5. This measurement may be obtained using incore or excore instrumentation. The second method involves interpolation between measured and predicted values. The nuclear design report provides predicted values for target flux difference at various cycle burnups. The difference between the last measured value and the predicted value at the same burnup is applied to the predicted value at the burnup where the target flux difference update is required. This revised predicted value can then be used to determine the updated value of the target flux difference.

SR 3-2-3-43 2-3-5

63 Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

63 A Frequency of 31 EFPD after each refueling and 92 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference to the value measured at steady state conditions. This is the basis for the CAOC. Remeasurement at this Surveillance interval also establishes the AFD target flux difference values that account for changes in incore-excore calibrations that may have occurred in the interim.

63 This SR 3-2-3-4 is modified by a Note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

(continued)

BASES

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
 2. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 3. UFSAR, Section 7.7.2.6.4.
-

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. Quadrant Power Tilt is a core tilt that is measured with the use of the excore power range flux detectors. A core tilt is defined as the ratio of maximum to average quadrant power. The QPTR is defined as the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants. Limiting the QPTR prevents radial xenon oscillations and will indicate any core asymmetries.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

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APPLICABLE
SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

- a. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F (Ref. 1);

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- c. During an ejected rod accident, the energy deposition to the fuel will be below 200 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$), and Bank Insertion, Sequence and Overlap Limits are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F_{\Delta H}^N$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F_{\Delta H}^N$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR monitor alarm shall be OPERABLE and QPTR shall be maintained at or below the limit of 1.02.

QPTR is monitored on an automatic basis using the Plant Process Computer System (PPCS) that has a QPTR monitor alarm. The PPCS determines from the excore detector outputs the ratio of the highest average nuclear power in any quadrant to the average of nuclear power in the four quadrants and provides an alarm message if the QPTR is above the 1.02 limit.

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.025 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F_{\Delta H}^N$ is possibly challenged. However, the additional QPTR of 0.005 is provided for margin in the LCO.

(continued)

BASES

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits assumed in the safety analyses.

Applicability in MODE 1 \leq 50% RTP and in other MODES is not required because there is neither sufficient stored energy in the fuel nor sufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F_{\Delta H}^N$ and $F_{\alpha}(Z)$ LCOs still apply below 50% RTP, but allow progressively higher peaking factors as THERMAL POWER decreases below 50% RTP.

ACTIONS

A.1

With the QPTR exceeding its limit, limiting THERMAL POWER to \geq 3% below RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. A further increase in the QPTR would require a lower limit to THERMAL POWER in accordance with Required Action A.2.

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A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR in accordance with SR 3.2.4.1 once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER must be limited accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

(continued)

BASES

ACTIONS
(continued)

A.3

The peaking factors $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of ~~within~~ 24 hours after achieving ~~equilibrium conditions with THERMAL POWER limited by~~ Required Actions A.1 and A.2 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F_{\Delta H}^N$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

68

The performance of SR 3.2.1.1 and SR 3.2.2.1 is no longer required once Condition A is exited. When the plant is already performing SR 3.2.1.2 or SR 3.2.2.2 to satisfy other requirements, SR 3.2.1.2 or SR 3.2.2.2 do not need to be suspended in order to perform SR 3.2.1.1 or SR 3.2.2.1 since the performance of SR 3.2.1.2 and SR 3.2.2.2 meet the requirements of SR 3.2.1.1 and SR 3.2.2.1, respectively.

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Insert next page →

A.4.5

If the QPTR has exceeded the 1.02 limit and the verification of $F_{\Delta H}^N$ and $F_Q(Z)$ shows that safety requirements are met, the excor detectors are recalibrated ~~normalized~~ to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to allow the operator to clearly detect any subsequent significant changes in QPTR and to provide a meaningful QPTR alarm.

62

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(continued)



A.4

Although $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.



the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions.

QPTR
B 3.2.4

BASES

ACTIONS

A.4⁵ (continued)

Required Action A.4⁵ is modified by a Note that states that the recalibration of the excore detectors cannot be performed ~~indicated tilt is not eliminated~~ until after there is verification that the hot channel factors are within limits (i.e., Required Action A.3⁵). It is necessary to verify that the core power distribution is acceptable prior to adjusting the excore detectors to show zero ~~eliminate the indicated~~ tilt and increasing power to ensure that the plant is not operating in an unanalyzed condition.

62

The Note is intended to prevent any ambiguity about the required sequence of actions.

A.5

After the flux tilt is zeroed out ~~normalized to eliminate the indicated tilt~~ (i.e., Required Action A.4 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.5 requires verification that $F_0(Z)$ and $F_{\Delta H}^N$ are within their specified limits within 24 hours of ~~after reaching RTP equilibrium conditions with THERMAL POWER increased above the limit of Required Actions A.~~ As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun and A.2. ~~These Completion Times are~~ time is intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

63

RTP.

STEP

(continued)



BASES

ACTIONS

A.5 (continued)

Required Action A.5 is modified by ~~three~~^{two} Notes. The first Note states that it is not necessary to perform Required Action A.5 if the cause of the QPTR alarm is associated with instrumentation alignment. The intent of this Note is to clarify that the core power distribution does not have to be re-verified if the QPTR alarm is only due to the instrumentation (i.e., the excore detectors) ~~being out of alignment~~ and not due to an anomaly within the core. The second Note states that the peaking factor surveillances are not required until after the excore detectors have been ~~calibrated~~^{normalized} to show zero ~~eliminate the indicated tilt~~ (i.e., Required Action A.4). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are adjusted to show zero ~~eliminate the indicated tilt~~ and the core returned to power.

(6B)

STET

~~The third Note states that only one of the following Completion Times, whichever becomes applicable first, must be met. The intent of this Note is to clearly indicate that the first Completion Time to become applicable is the Completion Time which must be met to satisfy Required Action A.5.~~

B.1

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If Required Actions A.1 through A,⁵ are not completed within their associated Completion Times, the plant must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to \leq 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

(continued)

BASES

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

When the QPTR monitor alarm is inoperable the QPTR must be verified within limits at a frequency of every 24 hours to ensure that the plant does not operate in an unanalyzed condition. When THERMAL POWER is $\geq 75\%$ RTP and one power range channel is inoperable, QPTR cannot be adequately measured using the excore detectors. In this situation a flux map must be completed to verify that the core power distribution is consistent with the safety analyses. A Completion Time of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt and provides sufficient time to stabilize the plant and perform a flux map when necessary.

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1

~~This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, The Completion Time of 24 hours is within its limits also consistent with the Frequency of SR 3.2.4.3 with one inoperable power range channel since these channels provide input into the QPTR monitor.~~

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1

This SR is the verification that the QPTR monitor is OPERABLE. This is normally accomplished by introducing a signal into the PPCS to verify control room annunciation of QPTR not within limit. The Frequency of 12 hours is sufficient to ensure OPERABILITY of the QPTR monitor since under normal plant operation, QPTR is not expected to significantly change.

SR 3.2.4.2

(continued)

BASES

(67)

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.13.2.4.2 (continued)

(67)

SR 3.2.4.13.2.4.2 is modified by two Notes. The first allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. The second Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is \geq 75% RTP and one power range channel is inoperable. The intent of this Note is to clarify that when one power range channel is inoperable and THERMAL POWER is \geq 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. AboveAt or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_{cF_{\alpha}(Z)}$ and $F_{\Delta H}^N$ remain within limits and the core power distribution is consistent with the safety analyses.

(169)

(169)

(67)

SR 3.2.4.23.2.4.3

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits when the QPTR alarm is inoperable. The Frequency of 24 hours is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.23.2.4.3 (continued)

(67)

SR ~~3.2.4.23.2.4.3~~ is modified by three Notes. The first Note states that the surveillance is only required to be performed if the QPTR monitor alarm is inoperable. This surveillance requires a more frequent verification that the QPTR is within limit since the monitor alarm is inoperable. The second Note allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable. The third Note states that SR 3.2.1.2 and SR 3.2.2.2 should be performed if THERMAL POWER is \geq 75% RTP and one power range channel is inoperable. The intent of this Note is clarify that when one power range channel is inoperable and THERMAL POWER is \geq 75% RTP, a full core flux map should be performed to verify the core power distribution instead of using the three OPERABLE power range channels to verify QPTR. Above ~~At~~ or above 75% RTP with one power range channel inoperable, QPTR monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing a full core flux map provides an accurate alternative means for ensuring that $F_{\Delta H}$ and $F_{\Delta H}^N$ remain within limits and the core power distribution is consistent with the safety analyses.

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REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Section 15.4.5.
3. Atomic Industrial Forum (AIF) GDC 29, Issued for comment July 10, 1967.

Revised per #222

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

ECO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

NOTE:
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more Functions with one channel inoperable.</p> <p><u>OR</u></p> <p>Two source range channels inoperable.</p>	<p>A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).</p>	<p>Immediately</p>
<p>B. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>B.1 Restore channel to OPERABLE status.</p>	<p>48 hours</p>

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required Action and associated Completion Time of Condition B not met.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Initiate action to fully insert all rods.</p> <p><u>AND</u></p> <p>C.3 Place Control Rod Drive System in a condition incapable of rod withdrawal.</p>	<p>6 hours</p> <p>6 hours</p> <p>7 hours</p>
<p>D. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>D.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.</p>	<p>6 hours</p>

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>H. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>H.1 Restore at least one channel to OPERABLE status upon discovery of two inoperable channels.</p> <p>AND</p> <p>H.2 Suspend operations involving positive reactivity additions.</p> <p>AND</p> <p>H.3 Restore channel to OPERABLE status.</p>	<p>1 hour from discovery of two inoperable channels</p> <p>Immediately</p> <p>48 hours</p>
<p>I. Required Action and associated Completion Time of Condition H not met.</p>	<p>I.1 Initiate action to fully insert all rods.</p> <p>AND</p> <p>I.2 Place the Control Rod Drive System in a condition incapable of rod withdrawal.</p>	<p>Immediately</p> <p>1 hour</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>J. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>J.1 Suspend operations involving positive reactivity additions.</p> <p>AND</p> <p>J.2 Perform SR 3.1.1.1.</p>	<p>Immediately</p> <p>12 hours</p> <p>AND</p> <p>Once per 12 hours thereafter</p>
<p>K. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>K.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.</p> <p>Place channel in trip.</p>	<p>6 hours</p>
<p>L. Required Action and associated Completion Time of Condition K not met.</p>	<p>L.1 Reduce THERMAL POWER to < 8.5% RTP.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>M. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>M.1. -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. Place channel in trip.</p>	<p>6 hours</p>
<p>N. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>N.1 Restore channel to OPERABLE status.</p>	<p>6 hours</p>
<p>O. Required Action and associated Completion Time of Condition M or N not met.</p>	<p>O.1 Reduce THERMAL POWER to < 50% RTP.</p>	<p>6 hours</p>
<p>P. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>P.1. -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. Place channel in trip.</p>	<p>6 hours</p>

(continued)



ACTIONS (continued)

<u>CONDITION</u>	<u>REQUIRED ACTION</u>	<u>COMPLETION TIME</u>
<p><u>Q:</u> Required Action and Associated Completion Time of Condition P not met.</p>	<p><u>Q:1</u> Reduce THERMAL POWER to < 50% RTP.</p> <p><u>AND</u></p> <p><u>Q:2.1</u> Verify Steam Dump System is OPERABLE.</p> <p><u>OR</u></p> <p><u>Q:2.2</u> Reduce THERMAL POWER to < 8% RTP.</p>	<p><u>6 hours</u></p> <p><u>7 hours</u></p> <p><u>7 hours</u></p>
<p><u>R:</u> As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p><u>R:1</u> -----NOTE----- The inoperable train may be bypassed for up to 4 hours for surveillance testing of the other train.</p> <p>Restore train to OPERABLE status.</p>	<p><u>6 hours</u></p>
<p><u>S:</u> As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p><u>S:1</u> Verify interlock is in required state for existing plant conditions.</p> <p><u>OR</u></p> <p><u>S:2</u> Declare associated RTS Function channel(s) inoperable.</p>	<p><u>1 hour</u></p> <p><u>1 hour</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>T. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>NOTES:</p> <p>1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE.</p> <p>2. One RTB may be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE.</p> <hr/> <p>T.1 Restore train to OPERABLE status.</p>	<p>1 hour</p>
<p>U. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>U.1 Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.</p> <p>AND</p> <p>U.2 Restore trip mechanism to OPERABLE status.</p>	<p>1 hour from discovery of two inoperable trip mechanisms</p> <p>48 hours</p>
<p>V. Required Action and associated Completion Time of Condition R, S, T, or U not met.</p>	<p>V.1 Be in MODE 3.</p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

<u>CONDITION</u>	<u>REQUIRED ACTION</u>	<u>COMPLETION TIME</u>
<p>W. As required by Required Action A.1 and referenced by Table 3.3.1-1.</p>	<p>W.1 Restore at least one trip mechanism to OPERABLE status upon discovery of two RTBs with inoperable trip mechanisms.</p> <p><u>AND</u></p> <p>W.2 Restore trip mechanism or train to OPERABLE status.</p>	<p>1 hour from discovery of two inoperable trip mechanisms</p> <p>48 hours</p>
<p>X. Required Action and associated Completion Time of Condition W not met.</p>	<p>X.1 Initiate action to fully insert all rods.</p> <p><u>AND</u></p> <p>X.2 Place the Control Rod Drive System in a Condition incapable of rod withdrawal.</p>	<p>Immediately</p> <p>1 hour</p>

SURVEILLANCE REQUIREMENTS

NOTE
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

<u>SURVEILLANCE</u>	<u>FREQUENCY</u>
<p>SR 3.3.1.1 Perform CHANNEL CHECK.</p>	<p>12 hours</p>
<p>SR 3.3.1.2 <u>NOTE</u> Required to be performed within 12 hours after THERMAL POWER is \geq 50% RTP.</p> <p>Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output and adjust if calorimetric power is $>$ 2% higher than indicated NIS power.</p>	<p>24 hours</p>
<p>SR 3.3.1.3 <u>NOTES</u></p> <p>1. Required to be performed within 7 days after THERMAL POWER is \geq 50% RTP but prior to exceeding 90% RTP following each refueling and if the Surveillance has not been performed within the last 31 EFPD.</p> <p>2. Performance of SR 3.3.1.6 satisfies this SR.</p> <p>Compare results of the incore detector measurements to NIS AFD and adjust if absolute difference is \geq 3%.</p>	<p>31 effective full power days (EFPD)</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<u>SURVEILLANCE</u>	<u>FREQUENCY</u>
<p>SR 3.3.1.4 Perform TADOT.</p>	<p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.5 Perform ACTUATION LOGIC TEST.</p>	<p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.6 -----NOTE----- Not required to be performed until 7 days after THERMAL POWER is \geq 50% RTP, but prior to exceeding 90% RTP following each refueling. ----- Calibrate excore channels to agree with incore detector measurements.</p>	<p>92 EFPD</p>
<p>SR 3.3.1.7 -----NOTE----- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3. ----- Perform COT.</p>	<p>92 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<u>SURVEILLANCE</u>	<u>FREQUENCY</u>
<p>SR 3.3.1.8 <u>NOTE</u></p> <ol style="list-style-type: none"> 1. Not required for power range and intermediate range instrumentation until 4 hours after reducing power < 6% RTP. 2. Not required for source range instrumentation until 4 hours after reducing power < 5E-11 amps. <p>Perform GOT.</p>	<p>92 days</p>
<p>SR 3.3.1.9 <u>NOTE</u></p> <p>Setpoint verification is not required.</p> <p>Perform TADOT.</p>	<p>92 days</p>
<p>SR 3.3.1.10 <u>NOTE</u></p> <p>Neutron detectors are excluded.</p> <p>Perform CHANNEL CALIBRATION.</p>	<p>24 months</p>
<p>SR 3.3.1.11 Perform TADOT.</p>	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.12 -----NOTE----- Setpoint verification is not required. ----- Perform TADOT.</p>	<p>Prior to reactor startup if not performed within previous 31 days</p>
<p>SR 3.3.1.13 Perform COT.</p>	<p>24 months</p>

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Reactor Trip	3(a), 4(b), 5(a)	2	B, C	SR 3.3.1.11	NA
2. Power Range Neutron Flux					
a. High	1, 2	4	D, G	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.10	109% RTP
b. Low	1(b), 2	4	D, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	25% RTP
3. Intermediate Range Neutron Flux	1(b), 2	2	E, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
4. Source Range Neutron Flux	2(c)	2	F, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	(d)
	3(a), 4(a), 5(a)	2	H, I	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	(d)
	3(e), 4(e), 5(e)	1	J	SR 3.3.1.1 SR 3.3.1.10	NA

(continued)

(a) With Control Rod Drive (CRD) System capable of rod withdrawal, or all rods not fully inserted.

(b) THERMAL POWER < 6% RTP

(c) Both Intermediate Range channels < 5E-11 nmps

(d) UFSAR Table 7.2.3

(e) With CRD System incapable of withdrawal and all rods fully inserted. In this condition, the Source Range Neutron Flux function does not provide a reactor trip, only indication.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
5. Overtemperature ΔT	1(f)	4	D/G	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 1 (page 3.3-18)
6. Overpower ΔT	1(f)	4	D/G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	Refer to Note 2 (page 3.3-19)
7. Pressurizer Pressure					
a. Low	1(f)	4	K/D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 1865 psig
b. High	1(f)	3	D/G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 2385 psig
8. Pressurizer Water Level - High	1(f)	3	D/G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 88%
9. Reactor Coolant Flow - Low					
a. Single Loop	1(g)	3 per loop	H/O	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 90%
b. Two Loops	1(h)	3 per loop	K/D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≥ 90%

(continued)

(f) THERMAL POWER ≥ 8.5% RTP

(g) THERMAL POWER ≥ 50% RTP

(h) THERMAL POWER ≥ 8.5% RTP and Reactor Coolant Flow - Low (Single Loop) trip function blocked



Table 3.3.1-1 (page 3 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	(g)	1 per RCP	N/O	SR 3.3.1.11	NA
b. Two Loops	(d)	1 per RCP	K/L	SR 3.3.1.11	NA
11. Undervoltage Bus 11A and 11B	(f)	2 per bus	K/L	SR 3.3.1.9 SR 3.3.1.10	(d)
12. Underfrequency Bus 11A and 11B	(f)	2 per bus	K/L	SR 3.3.1.9 SR 3.3.1.10	≥ 57.5 Hz
13. Steam Generator (SG) Water Level—Low Low	(f)	3 per SG	O/G	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\geq 16\%$
14. Turbine Trip					
a. Low Autostop Oil Pressure	(g)(k)	3	P/O	SR 3.3.1.10 SR 3.3.1.12	(d)
b. Turbine Stop Valve Closure	(d)(k)	2	P/O	SR 3.3.1.12	NA
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	(f)	2	R/V	SR 3.3.1.11	NA

(continued)

(d) UFSAR Table 7.2.3

(f) THERMAL POWER $\geq 8.5\%$ RTP

(g) THERMAL POWER $\geq 50\%$ RTP

(i) THERMAL POWER $\geq 8.5\%$ RTP and RCP Breaker Position (Single Loop) trip Function blocked

(j) THERMAL POWER $> 8\%$ RTP, and either no circulating water pump breakers closed, or condenser vacuum ≤ 20 "

(k) THERMAL POWER $\geq 50\%$ RTP, 1 of 2 circulating water pump breakers closed, and condenser vacuum > 20 "

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE CODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
16. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(c)	2	S/V	SR 3.3.1-10 SR 3.3.1-13	≥ 5E-11 amp
b. Low Power Reactor Trip Block, P-7	1(f)	4 (power range only)	S/V	SR 3.3.1-10 SR 3.3.1-13	≤ 8.5% RTP
c. Power Range Neutron Flux, P-8	1(g)	4	S/V	SR 3.3.1-10 SR 3.3.1-13	≤ 50% RTP
d. Power Range Neutron Flux, P-9	1(k)	4	S/V	SR 3.3.1-10 SR 3.3.1-13	≤ 50% RTP
e. Power Range Neutron Flux, P-10	1(j), 2	4	S/V	SR 3.3.1-10 SR 3.3.1-13	≥ 6% RTP
17. Reactor Trip Breakers	1, 2 3(a), 4(a), 5(a)	2 trains 2 trains	I/V W/X	SR 3.3.1-4 SR 3.3.1-4	NA NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2 3(a), 4(a), 5(b)	1 each per RTB 1 each per RTB	U/V W/X	SR 3.3.1-4 SR 3.3.1-4	NA NA
19. Automatic Trip Logic	1, 2 3(a), 4(a), 5(a)	2 trains 2 trains	R/V W/X	SR 3.3.1-5 SR 3.3.1-5	NA NA

(a) With CRD system capable of rod withdrawal or all rods not fully inserted.

(b) THERMAL POWER < 6% RTP.

(c) Both Intermediate Range channels < 5E-11 amps.

(f) THERMAL POWER ≥ 8.5% RTP.

(g) THERMAL POWER ≥ 50% RTP.

(j) THERMAL POWER > 8% RTP, and either no circulating water pump breakers closed, or condenser vacuum < 20".

(k) THERMAL POWER ≥ 50% RTP, 1 of 2 circulating water pump breakers closed, and condenser vacuum > 20".

(l) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Trip Setpoint is defined by:

$$\text{Overtemperature } \Delta T \approx \Delta T_0 \left[K_1 + K_2 (P - P') + K_3 (T - T') \right] \left[\frac{1 + r_1 s}{1 + r_2 s} \right] = f(\Delta I)$$

Where:

ΔT is measured RCS ΔT , °F.

ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec⁻¹.

T is the measured RCS average temperature, °F.

T' is the nominal T_{avg} at RTP, °F.

P is the measured pressurizer pressure, psig.

P' is the nominal RCS operating pressure, psig.

K_1 is the Overtemperature ΔT reactor trip setpoint, 1.20.

K_2 is the Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, 0.000900.

K_3 is the Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, 0.0209.

r_1 is the measured lead/lag time constant, 25 seconds.

r_2 is the measured lead/lag time constant, 5 seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$$f(\Delta I) = 0 \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

$$f(\Delta I) = 1.3 [(q_t - q_b) - 13] \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$



Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Trip Setpoint is defined by:

$$\text{Overpower } \Delta T \leq \Delta T_0 \left[K_4 - K_5 (T - T^2) - K_6 \left[\frac{\tau_3 s T}{\tau_3 s + 1} \right] - f(\Delta I) \right]$$

Where:

ΔT is measured RCS ΔT , °F.

ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec⁻¹.

T is the measured RCS average temperature, °F.

T^2 is the nominal T_{avg} at RTP, °F.

K_4 is the Overpower ΔT reactor trip setpoint, 1.077.

K_5 is the Overpower ΔT reactor trip heatup setpoint penalty coefficient which is:

0.0 for $T < T^2$ and;

0.0011 for $T \geq T^2$.

K_6 is the Overpower ΔT reactor trip thermal time delay setpoint penalty which is:

0.0262 for increasing T and;

0.00 for decreasing T .

τ_3 is the measured lead/lag time constant, 10 seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$$f(\Delta I) = 0 \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$

$$f(\Delta I) = 1.3 \left[(q_t - q_b) - 13 \right] \quad \text{when } q_t - q_b \text{ is } > +13\% \text{ RTP}$$



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3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

NOTE:
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel or train.	Immediately
B. As required by Required Action A.1 and referenced by Table 3.3.2-1.	B.1 Restore channel to OPERABLE status.	48 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced by Table 3.3.2-1.	D.1 Restore channel to OPERABLE status.	48 hours
E. As required by Required Action A.1 and referenced by Table 3.3.2-1.	E.1 Restore train to OPERABLE status.	6 hours
F. As required by Required Action A.1 and referenced by Table 3.3.2-1.	<p>F.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels.</p> <p>Place channel in trip.</p>	6 hours
G. Required Action and associated Completion Time of Condition D, E, or F not met.	<p>G.1 Be in MODE 3.</p> <p>AND</p> <p>G.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>
H. As required by Required Action A.1 and referenced by Table 3.3.2-1.	H.1 Restore channel to OPERABLE status.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action A.1 and referenced by Table 3.3.2-1.	I.1 Restore train to OPERABLE status.	6 hours
J. As required by Required Action A.1 and referenced by Table 3.3.2-1.	<p>J.1 ----- NOTE ----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels.</p> <p>Place channel in trip.</p>	6 hours
K. Required Action and associated Completion Time of Condition H, I, or J not met.	<p>K.1 Be in MODE 3.</p> <p>AND</p> <p>K.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
L. As required by Required Action A.1 and referenced by Table 3.3.2-1.	<p>L.1 ----- NOTE ----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of the other channels.</p> <p>Place channel in trip.</p>	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>M: Required Action and associated Completion Time of Condition L not met.</p>	<p>M:1 Be in MODE 3. <u>AND</u> M:2 Reduce pressurizer pressure to < 2000 psig.</p>	<p>6 hours 12 hours</p>
<p>N: As required by Required Action A.1 and referenced by Table 3.3.2-1.</p>	<p>N:1 Declare associated Auxiliary Feedwater pump inoperable and enter applicable condition(s) of LCO 3.7.5 "Auxiliary Feedwater (AFW) System."</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE	FREQUENCY
SR 3.3.2.1 Perform CHANNEL CHECK	12 hours
SR 3.3.2.2 Perform COT	92 days
SR 3.3.2.3 NOTE Verification of relay setpoints not required. Perform TADOT	92 days
SR 3.3.2.4 NOTE Verification of relay setpoints not required. Perform TADOT	24 months
SR 3.3.2.5 Perform CHANNEL CALIBRATION	24 months
SR 3.3.2.6 Verify the Pressurizer Pressure - Low and Steam Line Pressure - Low Functions are not bypassed when pressurizer pressure > 2000 psig.	24 months
SR 3.3.2.7 Perform ACTUATION LOGIC TEST	24 months

Table 3.3.2-1 (page 1 of 3)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE NODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3	2	OK	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	OK	SR 3.3.2.7	NA	NA
c. Containment Pressure - High	1,2,3,4	3	OK	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 6.0 psig	≤ 4.0 psig
d. Pressurizer Pressure - Low	1,2,3 (a)	3	EN	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5 SR 3.3.2.6	≥ 17.15 psig	≥ 1750 psig
e. Steam Line Pressure - Low	1,2,3 (a)	3 per steam line	EN	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5 SR 3.3.2.6	≥ 358 psig	≥ 514 psig
2. Containment Spray						
a. Manual Initiation						
Left pushbutton	1,2,3,4	1	OK	SR 3.3.2.4	NA	NA
Right pushbutton	1,2,3,4	1	OK	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	OK	SR 3.3.2.7	NA	NA
c. Containment Pressure - High High	1,2,3,4	3 per set	OK	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 32.5 psig	≤ 28 psig
3. Containment Isolation						
a. Manual Initiation	1,2,3,4	2	OK	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	OK	SR 3.3.2.7	NA	NA
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(a) Pressurizer Pressure = 2000 psig

Table 3.3.2-1 (page 2 of 3)
 Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation						
a. Manual Initiation	1, 2 (b), 3 (b)	1 per loop	D, G	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1, 2 (b), 3 (b)	2 trains	E, G	SR 3.3.2.7	NA	NA
c. Containment Pressure - High High	1, 2 (b), 3 (b)	3	F, G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 20 psig	≤ 18 psig
d. High Steam Flow	1, 2 (b), 3 (b)	2 per steam line	F, G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 0.55E6 lbm/hr @ 755 psig	≤ 0.4E6 lbm/hr @ 755 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with Low	1, 2 (b), 3 (b)	2 per loop	F, G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≥ 543°F	≥ 545°F
e. High-High Steam Flow	1, 2 (b), 3 (b)	2 per steam line	F, G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 3.7E6 lbm/hr @ 755 psig	≤ 3.6E6 lbm/hr @ 755 psig
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

(b) Except when both MSIVs are closed and deactivated.

Table 3.3.2-1 (page 3 of 3)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(c), 3(c)	2 trains	E/G	SR 3.3.2.7	NA	NA
b. SG Water Level - High	1,2(c), 3(c)	3 per SG	E/G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≤ 68%	≤ 67%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater (AFW)						
a. Manual Initiation						
AFW	1,2,3	1 per pump	N	SR 3.3.2.4	NA	NA
Standby AFW	1,2,3	1 per pump	N	SR 3.3.2.4	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	E/G	SR 3.3.2.7	NA	NA
c. SG Water Level - Low/Low	1,2,3	3 per SG	D/G	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.5	≥ 16%	≥ 17%
d. Safety Injection (Motor driven pumps only)	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
e. Undervoltage - Bus 11A and 11B (Turbine driven pump only)	1,2,3	2 per bus	D/G	SR 3.3.2.3 SR 3.3.2.5	≥ 2450 V With ≤ 3.6 sec time delay	≥ 2570 V With ≤ 3.6 sec time delay
f. Trip of Both Main Feedwater Pumps (Motor driven pumps only)	1,2	2 per HFW pump	B/C	SR 3.3.2.4	NA	NA

(c) Except when all Main Feedwater Regulating and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

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3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTES:
1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. NOTE: Not applicable to Functions 3 and 4.</p> <p>One or more Functions with one required channel inoperable.</p>	<p>A.1 Restore required channel to OPERABLE status.</p>	<p>30 days</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Initiate action to prepare and submit a special report.</p>	<p>Immediately</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. NOTE Only applicable to Functions 3 and 4. One or more Functions with required channel inoperable.</p>	<p>C.1 Restore required channel to OPERABLE status.</p>	<p>7 days</p>
<p>D. NOTE Not applicable to Function 11. One or more Functions with two required channels inoperable.</p>	<p>D.1 Restore one channel to OPERABLE status.</p>	<p>7 days</p>
<p>E. Two hydrogen monitor channels inoperable.</p>	<p>E.1 Restore one hydrogen monitor channel to OPERABLE status.</p>	<p>72 hours</p>
<p>F. Required Action and associated Completion Time of Condition C, D, or E not met.</p>	<p>F.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.</p>	<p>Immediately</p>
<p>G. As required by Required Action F.1 and referenced in Table 3.3.3-1.</p>	<p>G.1 Be in MODE 3. AND G.2 Be in MODE 4.</p>	<p>6 hours 12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action F.1 and referenced in Table 3.3.3-1	H.1 Initiate action to prepare and submit a special report.	Immediately

SURVEILLANCE REQUIREMENTS

NOTE
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation function in Table 3.3.3-1.

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2 Perform CHANNEL CALIBRATION.	24 months



Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION
1. Pressurizer Pressure	2	G
2. Pressurizer Level	2	G
3. Reactor Coolant System (RCS) Hot Leg Temperature	1 per loop	G
4. RCS Cold Leg Temperature	1 per loop	G
5. RCS Pressure (Wide Range)	2	G
6. RCS Subcooling Monitor	2	G
7. Reactor Vessel Water Level	2	H
8. Containment Sump B Water Level	2	G
9. Containment Pressure (Wide Range)	2	G
10. Containment Area Radiation (High Range)	2	H
11. Hydrogen Monitors	2	G
12. Condensate Storage Tank Level	2	G
13. Refueling Water Storage Tank Level	2	G
14. Residual Heat Removal Flow	2	G
15. Core Exit Temperature - Quadrant 1	2(a)	G
16. Core Exit Temperature - Quadrant 2	2(a)	G
17. Core Exit Temperature - Quadrant 3	2(a)	G
18. Core Exit Temperature - Quadrant 4	2(a)	G
19. Auxiliary Feedwater (AFW) Flow to Steam Generator (SG) A	2	G
20. AFW Flow to SG B	2	G
21. SG Water Level (Narrow Range) to SG A	2	G
22. SG Water Level (Narrow Range) to SG B	2	G

(continued)

(a) A channel consists of two core exit thermocouples (CETs).

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION
23. SG Water Level (Wide Range) to SG A	2	G
24. SG Water Level (Wide Range) to SG B	2	G
25. SG Pressure to SG A	2	G
26. SG Pressure to SG B	2	G



3.3 INSTRUMENTATION

3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.4 Each 480 V safeguards bus shall have two OPERABLE channels of LOP DG Start Instrumentation.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2,
"AC Sources - MODES 5 and 6."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each 480 V safeguards bus.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>216 A. One or more 480 V bus(es) with one channel inoperable.</p>	<p>A.1 Place channel(s) in trip.</p>	<p>6 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met. OR 216 TwoOne or more 480 V bus(es) with two channels per bus inoperable.</p>	<p>B.1 Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 When a channel is placed in an inoperable status solely for the performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 4 hours provided the second channel maintains LOP DG start capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform TADOT.	31 days
SR 3.3.4.2	Perform CHANNEL CALIBRATION with Trip Setpoint and Allowable Value for each 480 V bus as follows: a. Loss of voltage: Allowable Value Trip Setpoint Bus voltage ≥ 368 V ≥ 372.8 V Time delay ≤ 2.75 sec 2.4 ± 0.12 sec b. Degraded voltage: Allowable Value Trip Setpoint Bus voltage ≥ 414 V ≥ 419.2 V Time delay ≤ 1520 sec ≤ 1520 sec	24 months

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3.3 INSTRUMENTATION

~~3.3.5 Control Room Emergency Air Treatment System (CREATS) Actuation~~
~~Containment Ventilation Isolation Instrumentation~~

LC0 3.3.5 The ~~CREATS actuation~~ ~~Containment Ventilation Isolation~~ instrumentation for each function in ~~3.3.5-1~~ Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: ~~MODES 1, 2, 3, and 4;~~
~~During CORE ALTERATIONS;~~
~~During movement of irradiated fuel assemblies within~~
~~containment.~~

~~ACTIONS~~

NOTE
Separate Condition entry is allowed for each function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	4 hours

(continued)



CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><u>B.</u> <u>NOTE</u> Only applicable in MODE 1, 2, 3, or 4.</p> <p>One or more Functions with one or more manual or automatic actuation trains inoperable.</p> <p><u>OR</u></p> <p>Both radiation monitoring channels inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A not met.</p>	<p><u>B.1</u> Enter applicable Conditions and Required Actions of IC0 3.6.3, "Containment Isolation Boundaries," for containment mini- purge isolation valves made inoperable by isolation instrumentation.</p>	<p><u>Immediately</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><u>C</u> -----NOTE----- Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment.</p> <p>One or more Functions with one or more manual or automatic actuation trains inoperable.</p> <p><u>OR</u></p> <p>Both radiation monitoring channels inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time for Condition A not met.</p>	<p><u>C.1</u> Place and maintain containment purge and exhaust valves in closed position.</p> <p><u>OR</u></p> <p><u>C.2</u> Enter applicable Conditions and Required Actions of LCO 3.9.3, "Containment Penetrations," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.</p>	<p>Immediately</p> <p>Immediately</p>



SURVEILLANCE REQUIREMENTS

NOTE
Refer to Table 3.3.5-1 to determine which SRs apply for each Containment Ventilation Isolation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.5.1 Perform CHANNEL CHECK	24 hours
SR 3.3.5.2 Perform COT	92 days
SR 3.3.5.3 Perform ACTUATION LOGIC TEST	24 months
SR 3.3.5.4 NOTE Verification of setpoint is not required. Perform TADOT	24 months
SR 3.3.5.5 ⁴ Perform CHANNEL CALIBRATION	24 months

Containment Ventilation Isolation Instrumentation
3.3.5

Table 3.3.5-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1 Manual Initiation	2	SR 3.3.5.4	NA
2 ¹ Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.5.3	NA
2 ² Containment Radiation			
a. Gaseous	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
b. Particulate	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
3 ¹ Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," function 3, for all initiation functions and requirements.		
4 ¹ Containment Spray - Manual Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," function 2.g, for all initiation functions and requirements.		

Notes:

(a) Per Radiological Effluent Controls Program.

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

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LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME TIME TIME
<p>A. A One or more Functions with one or more channels inoperable.</p> <p>(151)</p>	<p>A.1 -----NOTE----- The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition. ----- Place CREATS in Mode F.</p>	<p>1 hour</p>
<p>B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.</p>	<p>6 hours 36 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1 Initiate action to restore channel(s) or train to OPERABLE status.	Immediately
	<u>AND</u>	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.3 Suspend movement of irradiated fuel assemblies.	Immediately



SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.5-13.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.5.13.3.6.1 Perform COT.	92 days
<p>-----NOTE----- SR 3.3.5.23.3.6.2 Verification of setpoint is not required. ----- Perform TADOT.</p>	24 months
SR 3.3.5.33.3.6.3 Perform CHANNEL CALIBRATION.	24 months
<p>Table 3.3.5-1 (page 1 of 1) CREATS Actuation Instrumentation</p> <p>SR 3.3.6.4 Perform ACTUATION LOGIC TEST.</p>	24 months



Table 3.3.6-1 (page 1 of 1)
CREATS Actuation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1 train	SR 3-3-5-23 3.6.2	NA
2. Automatic Actuation Logic and Actuation Relays	1 train	SR 3-3-5-13 3.6.4	NA
3. Control Room Radiation Intake Monitor			
a. Iodine	1	SR 3-3-5-13 3.6.1 SR 3-3-5-33 3.6.3	$\leq 9 \times 10^9$ ci/cc $\mu\text{Ci/cc}$
b. Noble Gas	1	SR 3-3-5-13 3.6.1 SR 3-3-5-33 3.6.3	$\leq 1 \times 10^6$ $\mu\text{Ci/cc}$
c. Particulate	1	SR 3-3-5-1 3-3-6-1 SR 3-3-5-33 3.6.3	$\leq 1 \times 10^9$ $\mu\text{Ci/cc}$



B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

Atomic Industry Forum (AIF) GDC 14 (Ref. 1) requires that the core protection systems, together with associated engineered safety features equipment, be designed to prevent or suppress conditions that could result in exceeding acceptable fuel design limits. The RTS initiates a plant shutdown, based on the values of selected plant parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (A00s) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The installed protection and monitoring systems have been designed to assure safe operation of the reactor at all times. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs with respect to these parameters and other reactor system parameters and equipment.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the associated LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs). These acceptable limits are:

- a. The Safety Limit (SL) values shall be maintained to prevent departure from nucleate boiling (DNB);
- b. Fuel centerline melt shall not occur; and
- c. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," maintains the above values and assures that offsite dose will be within 10 CFR 100 limits (Ref. 2) during A00s.

(continued)

BASES

BACKGROUND
(continued)

DBAs are events that are analyzed even though they are not expected to occur during the plant life. The DBA acceptance limit is that offsite doses shall be maintained within an acceptable fraction of 10 CFR 100 limits (Ref. 2). There are five different accident categories which are organized based on the probability of occurrence (Ref. 3). Each accident category is allowed a different fraction of the 10 CFR 100 limits, inversely proportioned to the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered as having acceptable consequences for that event.

The RTS instrumentation is segmented into three distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 4):

- a. Field transmitters or process sensors;
- b. Signal process control and protection equipment; and
- c. Reactor trip switchgear.

These modules are shown in Figure B 3.3.1-1 and discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. To account for the calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor as provided in established plant procedures.

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(continued)

BASES

BACKGROUND
(continued)

Signal Process Control and Protection Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 7 (Ref. 4), Chapter 6 (Ref. 5), and Chapter 15 (Ref. 6). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter has no measurable setpoint and is only used as an input to the protection circuits (e.g., manual trip functions) two channels with a one-out-of-two logic are sufficient. A third channel is not required since no surveillance testing is required during the time period in which the parameter is required.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 7).

(continued)

BASES

BACKGROUND

Signal Process Control and Protection Equipment (continued)

The two, three, and four process control channels discussed above all feed two logic trains. Figure B 3.3.1-1 shows a two-out-of-four logic function which provides input into two logic trains (Train A and B). Two logic trains are required to ensure that no single failure of one logic train will disable the RTS. Provisions to allow removing logic trains from service during maintenance are unnecessary because of the logic system's designed reliability. During normal operation, the two logic trains remain energized.

Reactor Trip Switchgear

The reactor trip switchgear includes the reactor trip breakers (RTBs) and bypass breakers as shown on Figure B 3.3.1-1. The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the control rod drive mechanisms (CRDMs). Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity and shutdown the reactor. Each RTB may be bypassed with a bypass breaker to allow testing of the RTB while the plant is at power. During normal operation, the output from the protection system is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the protection system output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open allowing the shutdown rods and control rods to fall into the core. Therefore, a loss of power to the protection system or RTBs will cause a reactor trip. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the protection system (except for the zirconium guide tube trip which only utilizes the undervoltage coils). Either the undervoltage coil or the shunt trip mechanism is sufficient by itself to open the RTBs, thus providing diverse trip mechanisms.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs which initiate in any MODE in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 6 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as ~~backups~~ ~~anticipatory trips~~ to RTS trip Functions that were credited in the accident analysis.

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The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for a RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO and
APPLICABILITY
(continued)

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The LCO and Applicability of each RTS Function are provided in Table 3.3.1-1. Included on Table 3.3.1-1 are Trip Setpoints for all applicable RTS Functions. Trip Setpoints for RTS Functions not specifically modeled in the safety analysis are based on established limits provided in plant procedures the UFSAR (Reference 4). Analytical values for RTS Functions which ensure that SLs are not violated during AOOs and that the consequences of DBAs will be acceptable, provided that the plant is operated within the LCOs, including any Required Actions that are in effect at the onset of the AOO or DBA and the equipment functions as designed are provided in plant procedures. Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS. The Trip Setpoints are the nominal limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy as specified within plant procedures. The channel containing the bistable is considered inoperable when the "as found" value exceeds the specified Trip Setpoint.

Specified in Table 3.3.3-1.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 4, 5, and 6. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.1-1 are therefore conservatively adjusted with respect to the analytical limits used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 8.

The RTS utilizes various permissive signals to ensure reactor trip Functions are in the correct configuration for the current plant status. These permissives back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Function is available.

(continued)

BASES

~~There are nine permissives in the RTS of which five are related to the APPLICABLE. The safety analyses and OPERABILITY requirements applicable MODES and specified conditions for RTS Functions specified~~

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~~SAFETY ANALYSES to each RTS Function and permissive provided in Table ECO, and 3.3.1-1 are discussed below:~~

~~APPLICABILITY
(continued)~~

- ~~1. —These five are discussed in detail below.~~

(continued)

BASES

~~APPLICABLE~~ P-6 Permissive

~~SAFETY ANALYSES,~~

~~LCO, and~~

~~APPLICABILITY~~

~~(continued)~~

The P-6 permissive permits bypassing the Source Range Neutron Flux trip Function during an approach to power. This permissive is derived from a bistable circuit in the Intermediate Range Neutron Flux instrumentation when any channel goes approximately one decade ($1E-10$ amps) above the minimum channel reading. After the permissive is effective, two defeat pushbuttons must be depressed to block the Source Range Neutron Flux trip Function. If both Intermediate Range Neutron Flux trip channels fall below $1E-10$ amps, the permissive is automatically defeated. The permissive may be manually defeated if below the P-10 setpoint by simultaneously depressing both defeat pushbuttons. During a decrease in power, the permissive resets at $5E-11$ amps.

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P-7 Permissive

The P-7 permissive is used to bypass the Pressurizer Pressure Low, Reactor Coolant, Flow Low (Two Loops), Reactor Coolant Pump (RCP) Breaker Position (Two Loops), and the Undervoltage Bus 11A and 11B trip Functions during low power and startup operations. The permissive is derived from a bistable circuit indicating $< 8.5\%$ RTP as measured by either the first stage turbine pressure or Power Range Neutron Flux instrumentation.

P-8 Permissive

The P-8 permissive allows a change in the Reactor Coolant Flow Low and RCP Breaker Position trip Functions so that a loss of a single loop will not cause a reactor trip. The permissive is set for $< 50\%$ RTP as sensed by the Power Range Neutron Flux instrumentation.

P-9 Permissive

The P-9 permissive prevents a reactor trip on Low Autostop oil pressure and Turbine Stop Valve Closure trip Functions when a turbine trip occurs $< 50\%$ RTP. This prevents unnecessary reactor trips when the steam dump system is available. The permissive receives input from condenser pressure and circulating water pump breaker position.

(continued)

BASES

~~APPLICABLE~~ P-10 Permissive

~~SAFETY ANALYSES,~~

~~LCO, and~~

~~APPLICABILITY~~

~~(continued)~~

The P-10 permissive is used to bypass the Intermediate Range Neutron Flux and Power Range Neutron Flux Low trip Functions during an approach to power. The permissive also provides a backup to the P-6 permissive to block the Source Range Neutron trip Function and provides input to the P-7 permissive. The permissive is derived from a bistable circuit indicating $> 8\%$ RTP as measured by the power range neutron flux instrumentation. In order to block these trip Functions, two pushbuttons for the Intermediate Range Flux trip Function and two pushbuttons for the Power Range Neutron Flux Low trip Function must be depressed after the permissive becomes effective. If THERMAL POWER is $\leq 8\%$ as measured by at least three out of four power range channels, the permissive automatically blocks these trip Functions. During a decrease in power, the permissive resets at $\leq 6\%$ RTP.

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The safety analyses and OPERABILITY requirements applicable to each RTS Function provided in Table 3.3.1-1 are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip Function ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip pushbuttons on the main control board. A Manual Reactor Trip energizes the shunt trip device and de-energizes the undervoltage coils for the RTBs and bypass breakers. It is used at the discretion of the control room operators to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint or during other degrading plant conditions.

(continued)

BASES

~~APPLICABLE~~ ~~1~~ ~~The LCO requires two Manual Reactor Trip channels to be~~
~~OPERABLE.~~ ~~Manual Reactor Trip (continued)~~

(222)
~~SAFETY ANALYSES,~~
~~LCO, and~~ ~~The LCO requires two Manual Reactor Trip channels to~~
~~APPLICABILITY~~ ~~be OPERABLE.~~ Each channel is controlled by a manual reactor trip pushbutton which actuates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single failure will disable the Manual Reactor Trip Function. This function has no adjustable trip setpoint with which to associate an LSSS, therefore no setpoints are provided.

(continued)



BASES

~~In MODE APPLICABLE 1 or 2, manual initiation capability of a reactor trip must be OPERABLE. Manual Reactor Trip (continued)~~

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SAFETY ANALYSES

LCO, and
APPLICABILITY

In MODE 1 or 2, manual initiation capability of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the RTBs are closed and the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip is not required to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods, or if one or more RTBs are open. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

2. Power Range Neutron Flux

The Power Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident. The Nuclear Instrumentation System (NIS) power range detectors (N-41, N-42, N-43, and N-44) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the CRD System for determination of automatic rod speed and direction. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

(LCO)

a. Power Range Neutron Flux-High Flux-High

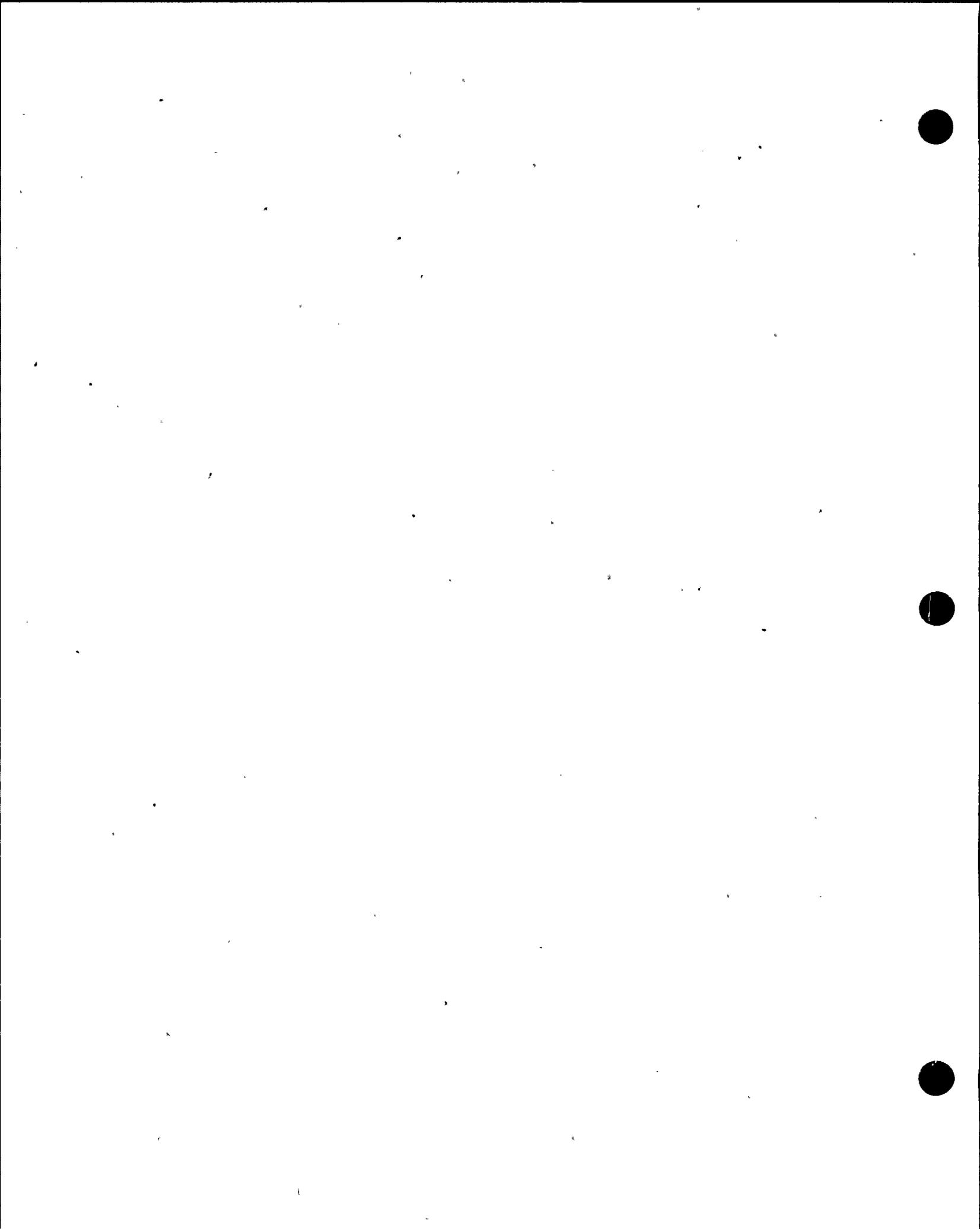
The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires all four of the Power Range Neutron Flux-High trip Function channels to be OPERABLE.

(continued)

BASES

(continued)



BASES

APPLICABLE

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LCO, and
APPLICABILITY

- a. Power Range Neutron Flux-High Flux-High
(continued)

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux-High trip Function is not required to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

- b. Power Range Neutron Flux-Low Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low trip Function channels (N-41, N-42, N-43, and N-44) to be OPERABLE.

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(continued)



BASES

APPLICABLE (169)

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LCO, and
APPLICABILITY

b. Power Range Neutron Flux-Low~~Flux-Low~~
(continued)

In MODE 1, below 6% RTP, and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than approximately 8% RTP (P-10 setpoint). This Function is automatically unblocked when three-out-of-four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function is not required to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux-Low trip Function and is not specifically modeled in the accident analysis. The NIS intermediate range detectors (N-35 and N-36) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

3. Intermediate Range Neutron Flux (continued)

The LCO requires two channels of the Intermediate Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. Because this trip Function is important only during low power conditions, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

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In MODE 1 below 6% RTP, and in MODE 2, the Intermediate Range Neutron Flux trip Function must be OPERABLE since there is a potential for an uncontrolled RCCA bank rod withdrawal accident. Above 8% RTP (P-10 setpoint), the Power Range Neutron Flux-High trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip Function is not required to be OPERABLE because the NIS intermediate range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against reactivity additions or power excursions in MODE 3, 4, 5, or 6.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

4. Source Range Neutron Flux

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The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition and provides protection against boron dilution and rod ejection events. This trip Function provides redundant protection to the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip Functions in ~~MODES 1 and MODE 2~~ and is not specifically credited in the accident analysis at these conditions. The NIS source range detectors (N-31 and N-32) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

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The LCO requires two channels of Source Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux trip Function to be OPERABLE in MODE 3, 4, or 5 with ~~RTBs open or the CRD System not capable of rod withdrawal and all rods fully inserted.~~ In this case, the source range Function is to provide control room indication. The outputs of the Function to RTS logic are not required to be OPERABLE when the ~~RTBs are open or the CRD system is not capable of rod withdrawal and all rods fully inserted.~~

The Source Range Neutron Flux Trip Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

(continued)



BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

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4. Source Range Neutron Flux (continued)

In MODE 2 when both intermediate range channels are < 5E-11 amps (below the P-6 setpoint), the Source Range Neutron Flux trip ~~Function~~ must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are manually de-energized by the operator and are inoperable.

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In MODE 3; 4, or 5 with the RTBs ~~closed and the CRD System capable of rod withdrawal~~ or all rods are not fully inserted, the Source Range Neutron Flux trip Function must be OPERABLE to provide core protection against a rod withdrawal accident. If the RTBs are open or the CRD System is not capable of rod withdrawal and all rods are fully inserted, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

(continued)

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SAFETY ANALYSES,
LCO, and
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(continued)

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit departure from nucleate boiling ratio (DNBR) is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, T_{avg} , axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow.

Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The ~~overtemperature~~ ~~Overtemperature~~ ~~Overtemperature~~ ΔT trip Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses the ΔT of each loop as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

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- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution $f(\Delta I)$ — the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

(continued)

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5. Overtemperature ΔT (continued)

The Overtemperature ΔT trip Function is calculated in two channels for each loop as described in Note 1 of Table 3.3.1-1. A reactor trip occurs if the Overtemperature ΔT Trip Setpoint is reached in two-out-of-four channels. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent an unnecessary reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

(continued)

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(continued)

6. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system; and
- axial power distribution $f(\Delta I)$ — the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

(continued)

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6. Overpower ΔT (continued)

The Overpower ΔT trip Function is calculated in two channels for each loop as described in Note 2 ~~to~~ of Table 3.3.1-1. A reactor trip occurs if the Overpower ΔT trip setpoint is reached in two-out-of-four channels. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent an unnecessary reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only MODES where enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

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(continued)

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(continued)

7. Pressurizer Pressure

The same sensors (PT-429, PT-430, and PT-431) provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip with the exception that the Pressurizer Pressure-Low and Overtemperature ΔT trips also receive input from PT-449. Since the Pressurizer Pressure channels are also used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function.

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a. Pressurizer Pressure-Low Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The LCO requires four channels of the Pressurizer Pressure-Low trip Function to be OPERABLE. Included within the four channels are lead time and lead/lag constraints.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip function must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (8.5% RTP). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, the Pressurizer Pressure-Low trip Function is not required to be OPERABLE because no conceivable power distributions can occur that would cause DNB concerns.

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(continued)

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(continued)

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions. The LCO requires three channels of the Pressurizer Pressure-High trip Function to be OPERABLE.

In MODE 1 or 2, the Pressurizer Pressure-High trip Function must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function is not required to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when in or below MODE 4.

¹⁰⁹ 8. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves.

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~~APPLICABLE~~ ~~8.~~ ~~Pressurizer Water Level High~~

~~SAFETY ANALYSES,~~

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~~(continued)~~

~~The Pressurizer Water Level High trip Function provides a backup signal for the Pressurizer Pressure High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. This trip Function is not specifically modeled in the accident analysis.~~

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The LCO requires three channels of the Pressurizer Water Level High trip Function to be OPERABLE APPLICABLE 8. Pressurizer Water Level-High

(continued)

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The LCO requires three channels of the Pressurizer Water Level-High trip Function to be OPERABLE. The pressurizer level channels (LT-426, LT-427, and LT-428) are also used for other control functions. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before the reactor high pressure trip.

In MODE 1 or 2, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip Function must be OPERABLE. In MODES 3, 4, 5, or 6, the Pressurizer Water Level-High trip Function is not required to be OPERABLE because transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

9. Reactor Coolant Flow-Low Flow-Low

The Reactor Coolant Flow-Low (Single Loop) and (Two Loops) trip Functions utilize three common flow transmitters per RCS loop to generate a reactor trip above 8.5% RTP (P-7 setpoint). Flow transmitters FT-411, FT-412, and FT-413 are used for RCS Loop A and FT-414, FT-415, and FT-416 are used for RCS Loop B.

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a. Reactor Coolant Flow-Low Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in the RCS loop, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, (50% RTP), a loss of flow in either RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

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The LCO requires three Reactor Coolant Flow-Low (Single Loop) trip Function channels per RCS loop to be OPERABLE in MODE 1 \geq 50% RTP (above P-8 setpoint). Each loop is considered a separate function for the purpose of this LCO.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint the Reactor Coolant Flow-Low (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

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b. Reactor Coolant Flow-Low Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in both RCS loops while avoiding reactor trips due to normal variations in loop flow.

The LCO requires three Reactor Coolant Flow-Low (Two Loops) trip Function channels per loop to be OPERABLE in MODE 1 above 8.5% RTP (P-7 setpoint) and before the Reactor Coolant Flow-Low (Single Loop) trip Function is OPERABLE (below the P-8 setpoint).

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~~APPLICABLE~~ Each loop is considered a separate function for the purpose of this LCO.

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~~Reactor Coolant Flow Low (Two Loops)~~
~~SAFETY ANALYSES, (continued)~~
~~LCO, and~~
~~APPLICABILITY~~

~~Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in both loops will initiate a reactor trip.~~

b. Reactor Coolant Flow Low (Two Loops)

~~SAFETY ANALYSES, (continued)~~
~~LCO, and~~
~~APPLICABILITY~~

~~Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in both loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.~~

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Below the P-7 setpoint, this trip Function is not required to be OPERABLE because all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip Function is not required to be OPERABLE because loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

10. RCP Breaker Position

Both RCP Breaker Position trip Functions (Single Loop and Two Loops) utilize a common auxiliary contact located on each RCP. These Functions anticipate the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates but are not specifically credited in the accident analysis.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of

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flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open ~~above~~ 50% RTP, a reactor trip is initiated. This trip function will generate a reactor trip before the Reactor Coolant Flow-Low (Single Loop) Trip Setpoint is reached.

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a. RCP Breaker Position (Single Loop) (continued)

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 above 50% RTP (above the P-8 setpoint). Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of plant SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip Function must be OPERABLE. In MODE 1 below the P-8 setpoint, the RCP Breaker Position (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

b. RCP Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops. The position of each RCP breaker is monitored. If both RCP breakers are open above 8.5% RTP (P-7 setpoint) and before the RCP Breaker Position ~~(One (Single Loop))~~ trip Function is OPERABLE (below the P-8 setpoint), a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached.

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LCO, and
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b. Reactor Coolant Pump Breaker Position (Two Loops)
(continued)

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 above the P-7 and below the P-8 setpoints. Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this Function because the RCS Flow-Low trip alone provides sufficient protection of plant SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow (including RCP breaker position) are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function is not required to be OPERABLE because a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

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SAFETY ANALYSES,
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11. Undervoltage-Bus Undervoltage-Bus 11A and 11B

The Undervoltage-Bus 11A and 11B reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops from a major network voltage distribution disturbance. The voltage to each RCP is monitored. Above 8.5% RTP (the P-7 setpoint), an undervoltage condition detected on both Buses 11A and 11B will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage Bus 11A and 11B channels to prevent reactor trips due to momentary electrical power transients.

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The LCO requires two Undervoltage-Bus 11A and 11B trip Function channels per bus to be OPERABLE in MODE 1 above the P-7 setpoint. Each bus is considered a separate function for the purpose of this LCO.

Below the P-7 setpoint, the Undervoltage-Bus 11A and 11B trip Function is not required to be OPERABLE because all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Undervoltage-Bus 11A and 11B is automatically enabled.

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LCO₇ and

APPLICABILITY
(continued)

12. ~~Steam Generator Water Level-Low-Low~~ Underfrequency - Bus
~~11A and 11B~~

The ~~Steam Generator (SG) Water Level-Low~~ Underfrequency - Bus 11A and 11B reactor trip Function ensures that protection is provided against a ~~loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to violating the DNBR limit due to uncovering the SG tubes~~ a loss of flow in both RCP loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above 8.5% RTP (the P-7 setpoint), a loss of frequency detected on both RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Underfrequency - Bus 11A and 11B channels per bus to be OPERABLE. Each bus is considered a separate Function for the purpose of this LCO.

Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Underfrequency - Bus 11A and 11B is automatically enabled.

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LCO, and
APPLICABILITY
(continued)

Steam Generator Water Level - Low Low

The Steam Generator (SG) Water Level - Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. Three level transmitters per SG (LT-461, LT-462, and LT-463 for SG A and, LT-471, LT-472, and LT-473 for SG B) provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the Engineered Safety Feature Actuation System (ESFAS) function of starting the AFW pumps on low low SG level. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor.

The LCO requires three trip Function channels of SG Water Level - Low Low per SG to be OPERABLE in MODES 1 and 2. Each SG is considered a separate Function for the purpose of this LCO.

In MODE 1 or 2, the SG Water Level - Low Low trip Function must be OPERABLE to ensure that a heat sink is available to the reactor. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low trip Function is not required to be OPERABLE because the reactor is not operating. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

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APPLICABLE ~~13.~~

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BASES

APPLICABLE 14. Turbine Trip
SAFETY ANALYSES,
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(continued)

Credit for these trip Functions is not credited in the accident analysis.

a. Turbine Trip-Low Trip-Low Autostop Oil Pressure

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The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above 50% RTP² (the P-9 setpoint). Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. Three pressure switches monitor the control oil pressure in the Autostop Oil System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three trip Function channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, the Turbine Trip-Low Autostop Oil Pressure trip Function is not required to be OPERABLE because load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore, there is no potential for a turbine trip.

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LCO, and
APPLICABILITY
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b. Turbine Trip-Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above ~~50% RTP~~ (The P-9 setpoint). Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If both limit switches indicate that the stop valves are closed, a reactor trip is initiated.

This Function only measures the discrete position (open or closed) of the turbine stop valves. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

The LCO requires two Turbine Trip-Turbine Stop Valve Closure trip Function channels, one per valve, to be OPERABLE in MODE 1 above P-9. Both channels must trip to cause reactor trip.

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b. Turbine Trip-Turbine Trip-Turbine Stop Valve Closure
(continued)

Below the P-9 setpoint, the Turbine Trip-Turbine Stop Valve Closure trip Function is not required to be OPERABLE because a load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore there is no potential for a turbine trip.

1415. Safety Injection Input from Engineered Safety Feature Actuation System

The Safety Injection (SI) Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This trip is assumed in the safety analyses for the loss of coolant accident (LOCA). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoints are not applicable to this Function. The SI Input is provided by relays in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trip Function channels of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

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Reactor Trip System Interlocks

SAFETY ANALYSES,

LCO, and
APPLICABILITY
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Reactor protection interlocks (i.e., permissives) are provided to ensure reactor trips are in the correct configuration for the current plant status. They back up operator actions to ensure protection system functions are not bypassed during plant conditions under which the safety analysis assumes the functions are not bypassed. Therefore, the interlock functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6 Permissive

The Intermediate Range Neutron Flux, P-6 permissive is actuated when any NIS intermediate range channel goes approximately one decade (10^1 E-10 amps) above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 permissive ensures that the following functions are performed:

• on increasing power, the P-6 interlock allows the manual block of the NIS Source Range Neutron Flux reactor trip by use of two defeat push buttons. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and

• on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the Source Range Neutron Flux reactor trip at $5E-11$ amps.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY a. Intermediate Range Neutron Flux, P-6 Permissive (continued)

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 permissive to be OPERABLE in MODE 2 when below the P-6 permissive setpoint.

Above the P-6 permissive setpoint, the Source Range Neutron Flux reactor trip will be blocked, and this function is no longer required.

In MODE 3, 4, 5, or 6 the P-6 permissive does not have to be OPERABLE because the Source Range is providing the required core protection.

b. Low Power Reactor Trips Block, P-7 Permissive

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The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or from first stage turbine pressure. The LCO requirement for the P-7 permissive allows the bypass of the following functions:

- Pressurizer Pressure - Low;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage - Bus 11A and 11B; and
- Underfrequency - Bus 11A and 11B.

These reactor trip functions are not required below the P-7 setpoint since the RCS is capable of providing sufficient natural circulation without any RCP running.

The LCO requires four channels of Low Power Reactor Trips Block, P-7 permissive to be OPERABLE in MODE 1 \geq 8.5% RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY b. Low Power Reactor Trips Block, P-7 Permissive (continued)

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the permissive performs its Function when power level drops below 8.5% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8 Permissive

The Power Range Neutron Flux, P-8, permissive is actuated at approximately 49% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock allows the Reactor Coolant Flow - Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on low flow in one or more RCS loops to be blocked so that a loss of a single loop will not cause a reactor trip. The LCO requirement for this trip Functions ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than 50% power.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1 $\geq 50\% RTP$

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 permissive must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

In MODE 1 $\geq 50\% RTP$, the Function is not required to be OPERABLE because a loss of flow in one loop will not result in DNB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

d. Power Range Neutron Flux, P-9 Permissive

The Power Range Neutron Flux, P-9 permissive is actuated at approximately 49% power as determined by two-out-of-four NIS power range detectors if the Steam Dump System is available and at 8% if the Steam Dump System is unavailable. The LCO requirement for this function ensures that the Turbine Trip - Low Autostop Oil Pressure and Turbine Trip - Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System and RCS. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

222

The LCO require four channels of Power Range Neutron Flux, P-9 permissive to be OPERABLE in MODE 1.

above the permissive setpoint

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System and RCS, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

MODE 2 below the permissive setpoint

e. Power Range Neutron Flux, P-10 Permissive

The Power Range Neutron Flux, P-10 permissive is actuated at approximately 8% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 8% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 permissive ensures that the following functions are performed:

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY e. Power Range Neutron Flux, P-10 Permissive (continued)

• on increasing power, the P-10 permissive allows the operator to manually block the Intermediate Range Neutron Flux and Power Range Neutron Flux-low reactor trips;

• on increasing power, the P-10 permissive automatically provides a backup signal to the P-6 permissive to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detector;

• the P-10 interlock provides one of the two inputs to the P-7 interlock; and

222

• on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODES 1 and 2.

< 6% R-TF and MODE

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power functions in the event of a reactor shutdown. This Function must also be OPERABLE in MODE 2 to ensure that core protection is providing during a startup or shutdown by the Power Range Neutron Flux-low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

< 6% R-TF

(continued)

BASES

~~THE FOLLOWING TEXT WAS MOVED
APPLICABLE 17.~~

~~THE PRECEDING TEXT WAS MOVED~~

Reactor Trip Breakers

SAFETY ANALYSES,

LCO, and

APPLICABILITY

(continued)

(167)

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The OPERABILITY requirement for the individual trip mechanisms is provided in Function 16 below. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the CRD System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the ~~RTBs or associated bypass breakers are closed, and the CRD System is capable of rod withdrawal and all rods are not fully inserted.~~

(222)

1618. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the CRD System, or declared inoperable under Function 15 above. (169) OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

(169)

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the ~~RTBs and associated bypass breakers are closed, and~~

(continued)



BASES

(2??)

the CRD System is capable of rod withdrawal and all rods are not fully inserted.

(continued)



BASES

(222) APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) 19 169

Automatic Trip Logic

The LCO requirement for the RTBs (Functions ¹⁷15 and ¹⁸16) and Automatic Trip Logic (Function ¹⁹17) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is also equipped with a redundant bypass breaker to allow testing of the trip breaker while the plant is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE trains ensures that failure of a single logic train will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs and associated bypass breakers are closed, and the CRD System is capable of rod withdrawal and all rods are not fully inserted.

(227)

The RTS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

(227)

~~The ACTIONS for each inoperable RTS Function are identified by a Note has been added to the ACTIONS to clarify the Conditions column application of Table 3.3.1-1 Completion Time rules. A Note has been added to the ACTIONS to clarify the application of Completion Time rules.~~ The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to analytical values specified in plant procedures, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that

(continued)

BASES

channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

(continued)

BASES

ACTIONS
(continued)

As shown on Figure B 3.3.1-1, the RTS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Trip Logic (Function 17 in Table 3.3.1-1). Therefore, a channel may be inoperable due to the failure of a field instrument or a bistable failure which affects one or both RTS trains that is comprised of the RTBs and Automatic Trip Logic Function. The only exception to this are the Manual Reactor Trip and SI Input from ESFAS trip Functions which are defined strictly on a train basis (i.e., failure of these Functions may only affect one RTS train).

A.1

Condition A applies to all RTS protection functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable or if both source range channels are inoperable. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

222

When the number of inoperable channels in a trip Function exceed those specified in all related Conditions associated with a trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if the trip Function is applicable in the current MODE of operation. This essentially applies to the loss of more than one channel of any RTS Function except with respect to Conditions G, H, and JH.

B.1,

~~Condition B applies to the Manual Reactor Trip Function in MODE 1 or 2, 2.1, and B.2.2~~

222

~~Condition B applies to the Manual Reactor Trip Function in MODE 1 or 2. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the required safety function.~~

and MODES 3, 4, and 5 with the CRO system capable of rod withdrawal or all rods not fully inserted

(continued)



BASES

■

(continued)

BASES

ACTIONS

B.1 (continued)

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

2.1, and BC.2.2 (continued)

~~The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.~~

~~If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the plant must be brought to a MODE in which the LCO does not apply and C. To achieve this status, the plant must be brought to at least MODE 3~~

227

~~If the Manual Reactor Trip Function cannot be restored to OPERABLE status within 6 additional hours (54 hours total time) the allowed 48 hour Completion Time of Condition B, the plant must be brought to a MODE in which the LCO does not apply. The 6 additional hours to reach MODE 3 from full power operation in an orderly manner and without challenging to achieve this status, the plant systems is reasonable, based on operating experience must be brought to at least MODE 3 within 6 hours, action must be initiated within 6 hours to ensure that all rods are fully inserted, and the Control Rod Drive System must be placed in a condition incapable of rod withdrawal within 7 hours. With The Completion Times provide adequate time to exit the plant in MODE 3, Condition B no longer applies and Condition C is entered of Applicability from full power operation in an orderly manner without challenging plant systems based on operating experience.~~

CD.1

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux - High

(continued)

BASES

- * Power Range Neutron Flux = Low;
- * Overtemperature ΔT ;
- * Overpower ΔT ;
- * Pressurizer Pressure = High;
- * Pressurizer Water Level = High; and-G
- * SG Water Level = Low Low.

222

(continued)

BASES

2-

~~Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal:~~

- ~~• Manual Reactor Trip;~~
- ~~• RTBs;~~
- ~~• RTB Undervoltage and Shunt Trip Mechanisms; and~~
- ~~• Automatic Trip Logic~~**ACTIONS**

D.

(continued)

222
With one channel or train inoperable, the inoperable channel or train channel must be restored to OPERABLE status or placed in the tripped condition within 486 hours. If the affected function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the RTBs must be opened within the next hour (49 hours total time). The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, these Functions are no longer required.

(continued)



BASES

ACTIONS ~~C.1 and C.2 (continued)~~

~~The Completion Times are reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.~~

~~D.1.1 and D.1.2~~

~~With one of four Power Range Neutron Flux High trip Function channels inoperable, the inoperable channel must be restored to OPERABLE status or THERMAL POWER must be reduced to < 75% RTP within 24 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, one quarter of the radial power distribution monitoring capability is lost.~~

~~In addition to reducing THERMAL POWER, a known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one out of three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.~~

~~D.2.1 and D.2.2~~

~~As an alternative to the above actions, a full core flux map may be performed within 24 hours and every 24 hours thereafter. The inoperable channel must also be placed in the tripped condition within 72 hours. Calculating a full core flux map every 24 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued plant operation at power levels > 75% RTP. The 24 hour Frequency is consistent with SR 3.2.1.2 and SR 3.2.2.2. In addition to performing the full core flux maps a known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one out of three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.~~

(continued)



BASES

ACTIONS ~~—————~~ D.3
~~(continued)~~

~~As an alternative to both above Actions, if the inoperable channel cannot be placed in the tripped condition within the specified Completion Time, the plant may be placed in a MODE where this LCO does not apply. To achieve this, 6 hours (78 hours total) are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.~~

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 12 hours while performing routine surveillance testing of other channels. This includes placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 12 hour time limit is consistent with Reference 9.~~

2??

E.1 and E.2

~~Condition E applies to the following reactor trip Functions:~~

- ~~• Power Range Neutron Flux Low;~~
- ~~• Overtemperature ΔT ;~~
- ~~• Overpower ΔT ;~~
- ~~• Pressurizer Pressure High;~~
- ~~• Pressurizer Water Level High; and~~
- ~~• SG Water Level Low Low.~~

(continued)



BASES

~~ACTIONS~~ ~~E.1 and E.2~~ (continued)

~~With one channel inoperable, the known inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition. For the Power Range Neutron Flux-High, Power Range Neutron Flux-Low, Overtemperature ΔI , and Overpower ΔI functions, this results in a one-out-of-three logic for actuation. For the Pressurizer Pressure-High, Pressure-High and Pressurizer Water Level-High Functions, this results in a one-out-of-two logic for actuation. For the SG Water Level-Low Low Function, this results in a one-out-of-two logic per each affected SG for actuation. The 72~~6~~ hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.~~

222

~~If the inoperable channel cannot be placed in the tripped condition within the specified Completion Time, the plant must be brought to the Required Actions have been modified by a MODE in which the LCO does not apply. Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing surveillance testing of other channels. To achieve this status, this includes placing the plant must be placed in MODE 3 within the next 6 hours (78 hours total time) inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The Completion Time of 6 This 4 hours is reasonable, based on operating experience, applied to place each of the plant in MODE 3 from full power in an orderly manner and without challenging plant systems OPERABLE channels.~~

remaining

~~For the SG Water Level-Low Low Functions, Condition E applies on a per SG basis. The 4 hour time limit is consistent with Reference 9.~~

(continued)

BASES

~~This allows one inoperable channel from each SG to be considered on a separate condition entry basis.~~ ACTIONS E.

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels and E. The 12 hour time limit is consistent with Reference 92.~~

(continued)

Condition E applies to the Intermediate Range Neutron Flux trip Function when THERMAL POWER is above the P-6 setpoint (5E-11 amp as derived from a bistable circuit of the intermediate range channels) and below the P-10 setpoint (6% RTP as derived from a bistable circuit of the Power Range channels) and one channel is inoperable.

222

(continued)

BASES

ACTIONS ~~F.1 and F.2~~
(continued)

272

~~Condition F applies to the Intermediate Range Neutron Flux trip Function when THERMAL POWER is above the P-6 setpoint (5E-11 amp as derived from a bistable circuit of the intermediate range channels) and below the P-10 setpoint (8% RTP as derived from a bistable circuit of the Power Range channels) and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs a monitoring and protection function. With one NIS intermediate range channel inoperable, 2 hours is allowed to either reduce THERMAL POWER below the P-6 setpoint or increase THERMAL POWER above the P-10 setpoint. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel inoperability does not result in reactor trip.~~

(continued)



BASES

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Required Action E.2 is modified by a Note which states that the option to increase THERMAL POWER is not allowed if both intermediate range channels are inoperable or if THERMAL POWER is $< 5E-11$ amps. This prevents the plant from increasing THERMAL POWER when the trip capability of the Intermediate Range Neutron Flux trip Function is not available or if the plant has not yet entered this trip Function's MODE of Applicability.

222

(continued)



BASES

ACTIONS
(continued)

G
P.1

or F

222

If the Required Actions of Condition D or E cannot be met within the specified Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without

challenging plant systems ACTIONS — G.1 and G.2
(continued)

~~Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels when THERMAL POWER is above the P-6 setpoint (5E 11 amps as derived from a bistable circuit of the intermediate range channels) and below the P-10 setpoint (8% RTP as derived from a bistable circuit of the power range channels). Required Actions specified in this Condition are only applicable when the inoperability of both channels do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs a monitoring and protection function. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase with no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within 2 hours. Below P-6, the Source Range Neutron Flux channels perform the required monitoring and protection functions. The Completion Time of 2 hours will allow for a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.~~

~~If both Intermediate Range Neutron Flux trip channels are inoperable, the source range channels may be used to determine that the plant is below the P-6 setpoint. This action places the plant below the MODE in which the P-6 interlock is required. Since the Intermediate Range Neutron trip Function is not credited in the accident analysis, loss of both channels is not a loss of safety Function.~~

(continued)

BASES

ACTIONS ~~_____~~ H-1
(continued)

~~Condition H applies to the Intermediate Range Neutron Flux trip Function when THERMAL POWER is below the P-6 setpoint (5E 11 amps as derived from a bistable circuit of the intermediate range channels) and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the required monitoring and protection functions. Therefore, the inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint at which point the NIS intermediate range channels provide the monitoring and protection function.~~

~~Since the Intermediate Range Neutron Flux trip Function is not credited in the accident analysis, loss of both channels is not a loss of safety Function.~~

???

I-1

~~Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint.~~

~~G-1 and G-2~~ F-1, F-2, and F-3

~~Condition G applies to the Source Range Neutron Flux trip Function when in MODE 2, below the P-6 setpoint. In this Condition, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall the RTBs and RTBBs must be suspended opened immediately. With the RTBs and RTBBs opened, the core is in a more stable condition.~~

~~With one channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation since with only one source range channel OPERABLE, core protection is severely reduced. The inoperable channel must also be returned within 48 hours.~~

(continued)

BASES

JH.1

~~Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, or in MODE 3, 4, or 5 with the RTBs closed and the GRD System capable of rod withdrawal. In these Conditions, the NIS source range performs the monitoring and protection functions.~~

222

Condition H applies to an inoperable source range channel in MODE 3, 4, or 5 with the GRD System capable of rod withdrawal or all rods not fully inserted. With both source range channels inoperable in this Condition, the RTBs must be opened immediately since a safety function has been lost. NIS source range performs the monitoring and protection functions. With the RTBs open two channels inoperable, the core is in a more stable condition and the plant enters Condition L at least one channel must be restored to OPERABLE status within 1 hour.

(continued)

BASES

~~ACTIONS~~ ~~K~~ The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this interval.

~~With one of the source range channels inoperable, operations involving positive reactivity additions must be suspended immediately and 48 hours is allowed to restore it to OPERABLE status.~~

(continued)

222

~~Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. In this Condition, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to OPERABLE status. If the channel cannot be returned to OPERABLE status, 1 additional hour (49 hours total time) is allowed to open the RTBs. The allowance of 48 hours to restore the channel to OPERABLE status, and the 1 additional hour (49 hours total time) to open the RTBs, is consistent with Reference 10.~~

L.1 and L.2

~~Condition L applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the RTBs open or the rod control system not capable of rod withdrawal. In this Condition, the NIS source range performs the monitoring and protection functions. In this Condition, the NIS source range performs the monitoring and protection functions. This will preclude any power circulation.~~

The suspension of positive reactivity additions

(continued)

BASES

With one of the source range channels inoperable, 48 hours is allowed to restore it to OPERABLE status.

ACTIONS

1. If the channel cannot be returned to OPERABLE status, 1 additional hour (49 hours total time) is allowed to open the RTBs and L.2

(continued)

If the Source Range trip Function cannot be restored to OPERABLE status within the required Completion Time of Condition H, the plant must be brought to a MODE in which the requirement does not apply. The allowance of 48 hours to restore the channel to OPERABLE status, and the 1 additional hour (49 hours total time) to open the RTBs, is consistent with Reference 10.

L.1 and L.2

Condition L applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the RTBs open or the rod control system not capable of rod withdrawal. In this Condition, the NIS source range performs the monitoring and protection functions. To achieve this status, action must be immediately initiated to fully insert all rods. Additionally, the CRD System must be placed in a condition incapable of rod withdrawal within 1 hour. The Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event occurring during this interval.

J.1

Condition J applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods are fully inserted. In this Condition, the NIS source range performs the monitoring function. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

Also, the SDM must be verified once every within 12 hours and every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core

(continued)

BASES

222 protection is severely reduced. Verifying the SDM once per 12 hours allows sufficient time to perform the calculations and determine that the SDM requirements are met and to ensure that the core reactivity has not changed. Required Action 4.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Time of once per 12 hours is based on operating experience in performing the Required Actions and the knowledge that plant conditions will change slowly.

(continued)

BASES

ACTIONS
(continued)

~~MK.1 and M~~

~~Condition K applies to the following reactor trip Functions:~~

- ~~• Pressurizer Pressure - Low;~~
- ~~• Reactor Coolant Flow - Low (Two Loops);~~
- ~~• RCP Breaker Position (Two Loops);~~
- ~~• Undervoltage - Bus 11A and 11B; and~~
- ~~• Underfrequency - Bus 11A and 11B.~~

~~2~~

(continued)

~~Condition M applies. With one channel inoperable, the inoperable channel must be restored to the following reactor trip Functions:~~

- ~~• Pressurizer Pressure Low;~~
- ~~• Reactor Coolant Flow Low (Two Loops);~~
- ~~• RCP Breaker Position (Two Loops);~~
- ~~• Undervoltage - Bus 11A and 11B OPERABLE status or placed in the tripped condition within 6 hours.~~

~~With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 72 hours allowed to place the channel in the tripped condition is consistent with Reference 9 if the inoperable channel cannot be restored to OPERABLE status.~~

~~An additional 6 hours (78 hours total time) is allowed to reduce THERMAL POWER to < 8.5% RTP (P 7 setpoint) at which point the Function is no longer required. Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel(s), and the low probability of occurrence of an~~

(continued)

BASES

~~event during this period that may require the protection afforded by the Functions associated with Condition K.—~~

~~This places the plant in For the Reactor Coolant Flow—Low (Two Loops) Function, Condition K applies on a MODE where the LCO is no longer applicable per loop basis. An alternative is not provided for increasing THERMAL POWER above the P-8 setpoint for the Reactor Coolant Flow Low (Two Loops) and For the RCP Breaker Position (Two Loops) trip Functions since this places the plant in Condition N and Condition O Function, respectively Condition K applies on a per RCP basis.~~

222

~~Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel(s), For Undervoltage—Bus 11A and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with 11B, Condition—K applies on a per bus basis.~~

and Underfrequency Bus 11A and 11B

(continued)

BASES

ACTIONS ~~M.1 and M.2~~ (continued)

~~For the Reactor Coolant Flow Low (Two Loops) Functions, Condition M applies on a per loop basis. For the RCP Breaker Position (Two Loops) Function, Condition M applies on a per RCP basis. For Undervoltage Bus 11A and 11B, Condition M applies on a per bus basis. This allows one inoperable channel from each loop, RCP, or bus to be considered on a separate condition entry basis.~~

(222)

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 124 hours while performing routine surveillance testing of the other channels. The 124 hour time limit is consistent with Reference 9. The 4 hours is applied to each of the remaining OPERABLE channels.

(continued)

BASES

ACTIONS [L.1]
(continued)

(M) If the Required Action and Completion Time of Condition K is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 1 < 8.5% RTP (P-7 setpoint) at which point the Function is no longer required. An alternative is not provided for increasing THERMAL POWER above the P-8 setpoint for the Reactor Coolant Flow-Low (Two Loops) and RCP Breaker Position (Two Loops) trip Functions since this places the plant in Condition K². The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging plant systems.

M.1

(22?)

Condition M applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel per loop inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status or place in trip is consistent with Reference 9. The 4 hours is applied to each of the two OPERABLE channels.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hour time limit is consistent with Reference 9.

N.1-and-

Condition N applies to the RCP Breaker Position (Single Loop) trip Function.²

~~Condition N applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel per loop inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 72 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within 72 hours, then~~

(continued)

BASES

~~THERMAL POWER must be reduced to $< 50\%$ RTP (P 8 setpoint) within the next 6 hours (78 hours total time). This places the plant in a MODE where the LCO is no longer applicable. This trip Function is not required to be OPERABLE below 50% RTP because other RTS trip Functions can provide the necessary core protection. The 72 hours allowed to restore the channel to OPERABLE status or place in trip and the 6 additional hours allowed to reduce THERMAL POWER to $< 50\%$ RTP are consistent with Reference 9.~~

222

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is consistent with Reference 9.~~

(continued)

BASES

~~ACTIONS~~ 0.1 and 0.2
~~(continued)~~

222

~~Condition 0 applies to the RCP Breaker Position (Single Loop) trip Function. There is one breaker position device per RCP breaker. With one channel per RCP inoperable, the inoperable channel must be restored to OPERABLE status within 72 hours. If the channel cannot be restored to OPERABLE status within the 72 hours, then THERMAL POWER must be reduced to < 50% RTP (P 8 setpoint) within the next 6 hours (78 hours total time) allowed to restore the channel to OPERABLE status is consistent with Reference 9.~~

(continued)

BASES

~~This places the plant in a MODE where the LCO is no longer applicable~~
~~0. This Function~~

(continued)

~~If the Required Action and associated Completion Time of Condition M or N is not required to be OPERABLE below 50% RTP because other RTS met, the plant must be placed in a MODE where the Functions provide core protection are not required to be OPERABLE. The 72 hours allowed to restore the channel to OPERABLE to achieve this status and the 6 additional hours allowed to reduce THERMAL POWER must be reduced to < 50% RTP are consistent with Reference 9 (P-8 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.~~

P.1 and-

222

the P-4
a tripped

~~Condition P applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure in MODE 1 above 50% RTP.~~ 2

~~Condition P applies With one channel inoperable, the inoperable channel must be restored to Turbine Trip on Low Autostop Oil Pressure OPERABLE status or on Turbine Stop Valve Closure placed in MODE 1 above 50% RTP the tripped condition within 6 hours. With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped Condition, this results in a partial trip condition within 72 hours requiring only one additional channel to initiate a reactor trip. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel. The 6 hours allowed to initiate a reactor trip place the inoperable channel in the tripped condition is consistent with Reference 9. If the inoperable channel cannot be restored to OPERABLE status or placed in the tripped condition within 72 hours, then power must be reduced to < 50% RTP (P-9 setpoint) within the next 6 hours (78 hours total time) channel.~~

~~The 72 hours allowed to place Required Actions have been modified by a Note that allows placing the inoperable channel in the tripped bypassed condition and the 6 additional hours allowed for reducing power is consistent~~

(continued)

BASES

~~with Reference 9 up to 4 hours while performing surveillance testing of the other channels.~~

The 4 hour is applied to 2.5 h of the remaining OPERABLE channels.

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. A 4 hour time limit is consistent with Reference 9.~~

222

The 12 hour time limit is consistent with Reference 9.

(continued)

BASES

I, Q.2.1, and Q.2.2

227

If the Required Action and Associated Completion Time of Condition P are not met, the plant must be placed in a MODE where the Turbine Trip Functions are no longer required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-9 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

(continued)

BASES

ACTIONS
(continued)

Q.1 and Q.2

~~Condition 2.1 and Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. With one train inoperable, 6 hours are allowed to restore the train to 2.2 (continued)~~

222

~~The Steam Dump system must also be verified OPERABLE status within 7 hours or THERMAL POWER must be reduced to < 8% RTP. If this ensures that either the inoperable train cannot be restored to OPERABLE status within 6 hours, then secondary system or RCS is capable of handling the plant must be placed in heat rejection following a mode in which the LCO no longer applies reactor trip. This is accomplished by placing the plant in MODE 3 within need to perform the next 6 hours (12 hours total time) actions in an orderly manner and the low probability of an event occurring in this time. The Completion Time of 6 hours to restore the train to OPERABLE status (Required Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours to place the plant in MODE 3 (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.~~

~~The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.~~

R.1 and

~~Condition R applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. 2~~

~~Condition R applies to With one train inoperable, 6 hours is allowed to restore the RTBs in MODES 1 and 2 train to OPERABLE status. With one train inoperable, 1 hour is allowed The Completion Time of 6 hours to restore the train to OPERABLE status or the plant must be brought to a MODE's reasonable considering that in which the LCO does not apply this Condition, the remaining OPERABLE train is~~

(continued)



BASES

adequate to perform the safety function and given the low probability of an event during this interval.

~~To achieve this status~~The Required Action has been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, the plant must be placed in MODE 3 within the next 6 hours (7 hours total time) provided the other train is OPERABLE.—

222

~~The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the plant in MODE 3 removes the requirement for this particular function.~~

(continued)

BASES

ACTIONS ~~R.1 and R.2 (continued)~~

~~The Required Actions have been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 8 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE.~~

S.1 and S.2

(222)

~~Condition S applies to the RTB Undervoltage and Shunt Trip Mechanisms in MODES 1 and 2P-6, P-7, P-8, P-9, and P-10 permissives. With one trip mechanism for one RTB channel inoperable, it must be restored the associated interlock must be verified to an OPERABLE status within 48 hours or the plant must be placed in a MODE where the requirement does not apply its required state for the existing plant condition within 1 hour or the associated RTS channel(s) must be declared inoperable. This/These actions are conservative for the case where power level is accomplished by placing the plant in MODE 3 within the next 6 hours (54 hours total time) being raised. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. Verifying the interlock status manually accomplishes the interlock's Function. With the plant in MODE 3, Condition S no longer applies and Condition C. The Completion Time of 1 hour is entered based on operating experience and the minimum amount of time allowed for manual operator actions.~~

(continued)



BASES

ACTIONS
(continued)

Condition T applies to the RTBs in MODES 1 and 2. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status. The 1 hour Completion Time is based on operating experience and the minimum amount of time allowed for manual operator actions.

The Required Action has been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE.

222

U:1 and U:2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms (i.e., diverse trip features) in MODES 1 and 2. With two diverse trip features inoperable, at least one diverse trip feature must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism for one RTB inoperable, it must be restored to an OPERABLE status within 48 hours. The affected RTB shall not be bypassed while one of the diverse trip features is inoperable except for the time required to perform maintenance to one of the diverse trip features. The allowable time for performing maintenance of the diverse trip features is 86 hours for the reasons stated under Condition R.

The Completion Time of 48 hours for Required Action SU.12 is reasonable considering that in this Condition there is one remaining diverse trip feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

(continued)

BASES

(continued)

BASES

~~SURVEILLANCE~~ The SRs for each RTS Function are identified by the SRs
~~REQUIREMENTS~~ column of Table 3.3.1-1 for that Function. **ACTIONS**

V.1

(continued)

If the Required Action and Associated Completion Time of Condition R, S, I, or U is not met, the plant must be placed in a MODE where the functions are no longer required to be OPERABLE. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

W.1 and W.2

Condition W applies to the following reactor trip Functions in MODE 3, 4, or 5 with the CRD System capable of withdrawal or all rods not fully inserted:

- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

With two trip mechanisms inoperable, at least one trip mechanism must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

trip - With one trip mechanism or train inoperable, the inoperable two mechanism or train must be restored to OPERABLE status within 48 hours.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

(continued)

BASES

ACTIONS X.1 and X.2
(continued)

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action be must initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

222

SURVEILLANCE REQUIREMENTS The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies (Ref. 8).

SR 3.3.1.1

A CHANNEL CHECK is required for the following RTS trip functions:

- Power Range Neutron Flux - High;
- Power Range Neutron Flux - Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT ;

(continued)



BASES

772

- ~~Overpower ΔT ;~~
- ~~Pressurizer Pressure Low;~~
- ~~Pressurizer Pressure High;~~
- ~~Pressurizer Water Level High;~~
- ~~Reactor Coolant Flow Low (Single Loop);~~
- ~~Reactor Coolant Flow Low (Two Loops); and~~
- ~~SG Water Level Low Low~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1 (continued)

???

- Overpower ΔT ;
- Pressurizer Pressure - Low;
- Pressurizer Pressure - High;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Single Loop);
- Reactor Coolant Flow - Low (Two Loops); and
- SG Water Level - Low/Low

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel check acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

(continued)

BASES

222

SURVEILLANCE SR 3.3.1.2

(continued)

REQUIREMENTS

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux-High channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.3.1.2~~ (continued)
~~REQUIREMENTS~~

(169)

~~This SR 3.3.1.2 is modified by two Notes. Note which states that this Surveillance is required only if reactor power is $\geq 50\%$ RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. Note 1 indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is $> 2\%$ RTP. Note 2 clarifies that this Surveillance is required only if reactor power is $\geq 50\%$ RTP and that 12 hours is allowed for performing the first Surveillance after reaching 50% RTP. At lower power levels, calorimetric data are inaccurate.~~

(222)

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

This SR compares the incore system to the NIS channel output every 31 effective full power days (EFPD). If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the ~~Overtemperature~~ ~~Overpower~~ ~~ΔT Function and ΔT Functions.~~

(222)

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.3 (continued)

This SR is modified by ~~three~~^{two} Notes. Note 1 indicates ~~clarifies~~ that the ~~excure NIS channel~~ shall ~~Surveillance is required to be adjusted if the absolute difference between the incore performed within 7 days after THERMAL POWER is \geq 50% RTP but prior to exceeding 90% RTP following each refueling and excure AFD is \geq 3% if it has not been performed within the last 31 EFPD.~~ Note 2 ~~clarifies~~^{states} that the ~~Surveillance is required to be performed within 7 days after THERMAL POWER is \geq 50% RTP but prior to exceeding 90% RTP following each refueling and~~ ~~if performance of SR 3.3.1.6 satisfies this SR since it has not been performed within the last 31 EFPD is a more comprehensive test.~~ Note 3 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

(222)

The Frequency of every 31 EFPD is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

This SR is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS of the RTB, and the RTB Undervoltage and Shunt Trip Mechanisms. This test shall verify OPERABILITY by actuation of the end devices.

The test shall include separate verification of the undervoltage and shunt trip mechanisms except for the bypass breakers which do not require separate verification since no capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR ~~3.3.1.14~~^{3.3.1.11}. However, the bypass breaker test shall include a local shunt trip. ~~A Note has been added to indicate that this~~ ~~This~~ test must be performed on the bypass breaker prior to placing it in service to take the place of a RTP.

(222)

(continued)

BASES

The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.5

This SR is the performance of an ACTUATION LOGIC TEST on the RTS Automatic Trip Logic every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

This SR is a calibration of the excore channels to the incore channels every 92 EFPD. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the ~~overttemperature~~ Overtemperature ΔT Function.

This ~~SR 3.3.1.6~~ has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is $\geq 50\%$ RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 92 EFPD.

The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.7

~~SR 3.3.1.7~~ This SR is the performance of a COT every 92 days for the following RTS functions:

- Power Range Neutron Flux - High;
- Source Range Neutron Flux (in MODE 3, 4, or 5 with RTBs closed and the CRD System capable of rod withdrawal) ~~withdrawal or all rods not fully inserted~~);
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure - Low;
- Pressurizer Pressurizer - High;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Single Loop);
- Reactor Coolant Flow - Low (Two Loops); and
- SG Water Level - Low Low

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the analytical values specified in plant procedures. ~~Trip Setpoint of Table 3.3.1-1.~~ The "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.7 (continued)

(169)

~~SR 3.3.1.7~~² is modified by a Note that provides a 4 hour delay in the requirement to perform this surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the plant is in MODE 3 with the RTBs closed for greater than 4 hours, this SR must be performed within 4 hours after entry into MODE 3.

The Frequency of 92 days is consistent with Reference 109.

SR 3.3.1.8

(169)

This SR is the performance of a COT as described in SR 3.3.1.7 for the Power Range Neutron Flux - Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux (MODE 2), except that this test also includes verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition. ~~SR 3.3.1.8~~² is modified by two Notes that provide a 4 hour delay in the requirement to perform this surveillance. These Notes allow a normal shutdown to be completed and the plant removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days ~~hereafter~~ applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 4 hours after reducing power below P-10 or P-6.

(169)

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.8 (continued)

The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the Source range channels. Once the plant is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the plant in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

This SR is the performance of a TADOT for the Undervoltage - Bus 11A and 11B and Underfrequency - Bus 11A and 11B trip Functions. The Frequency of every 92 days is consistent with Reference 109.

222

This ~~SR 3.3.1.9~~ is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to Bus 11A and 11B undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION required by SR 3.3.1.10.

SR 3.3.1.10

This SR is the performance of a CHANNEL CALIBRATION for the following RTS Functions:

161

- Power Range Neutron Flux; ^{- High;}
- Power Range Neutron Flux - Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;

(continued)

D1

BASES

???

◆—Overtemperature ΔT ;

◆—Overpower ΔT ;

L

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3:1.10 (continued)

- ~~Overtemperature ΔT ;~~
- ~~Overpower ΔT ;~~
- Pressurizer Pressure - Low;
- Pressurizer Pressure - High;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Single Loop);
- Reactor Coolant Flow - Low (Two Loops);
- ~~Undervoltage - Bus~~ ~~Undervoltage - Bus~~ 11A and 11B;
- ~~Underfrequency - Bus 11A and 11B;~~
- SG Water Level - Low Low; and
- Turbine Trip - Low Autostop Oil Pressure; ~~and~~
- ~~Reactor Trip System Interlocks.~~

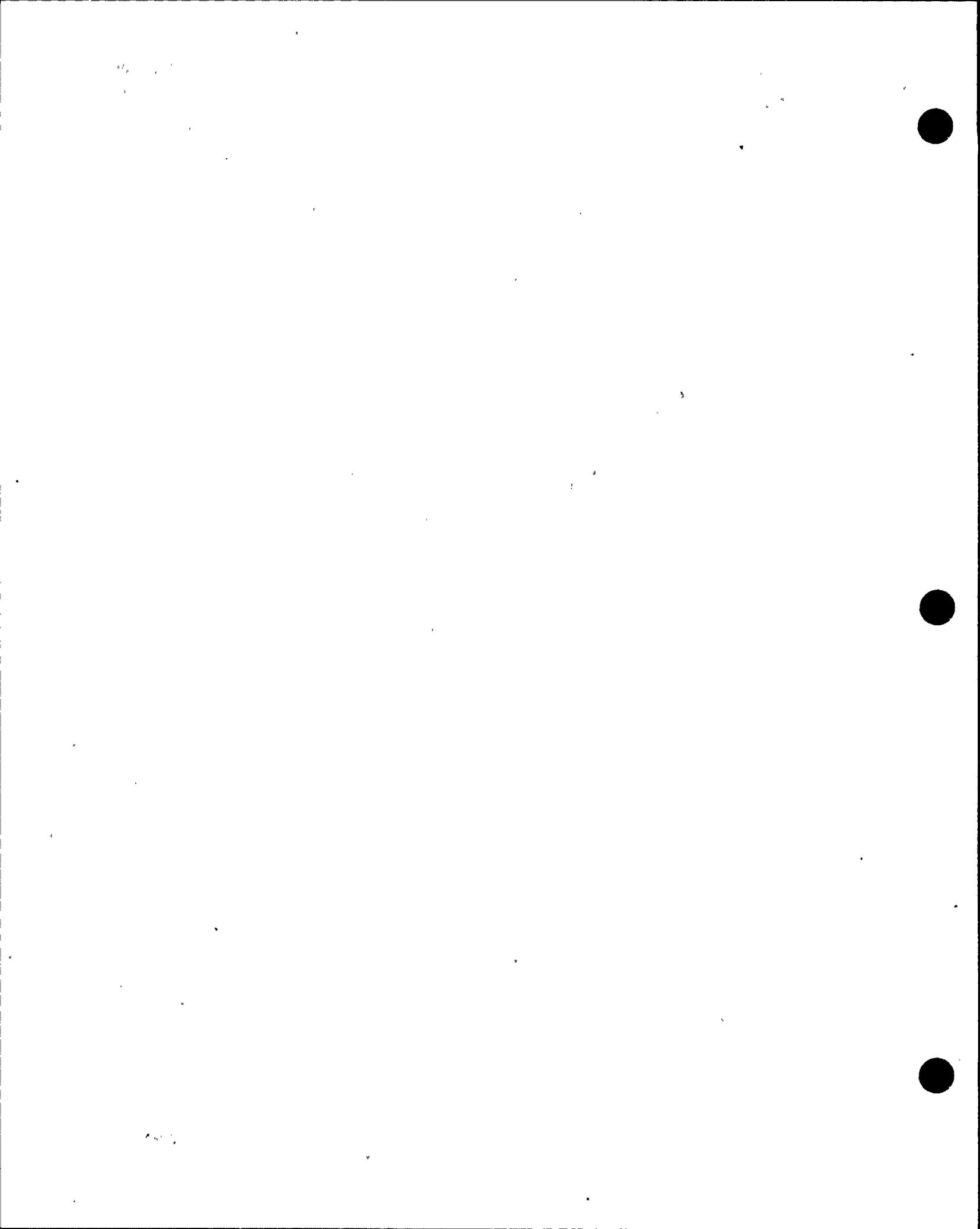
222

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology (Ref. 8). The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of 24 month calibration intervals in the determination of the magnitude of equipment drift in the setpoint methodology.

Whenever a sensing element is replaced, the next CHANNEL CALIBRATION of the resistance temperature detector (RTD) sensors shall include an in place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an in place cross calibration that compares the other (continued) sensing elements with the recently installed sensing element.



BASES

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.10 (continued)

~~SR 3.3.1.10~~ is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 50% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 24 month Frequency.

(continued)

BASES

SURVEILLANCE — SR 3.3.1.11

REQUIREMENTS

(continued)

No change
SR

This SR is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS trip Functions, ~~and the interlock Functions.~~ This TADOT is performed every 24 months. This test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. ~~The Reactor Trip Bypass Breaker test shall include testing of the undervoltage trip.~~

(69)

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT because the Functions affected have no setpoints associated with them.

(continued)

BASES

SURVEILLANCE SR 3.3.1.12

REQUIREMENTS

(continued)

This SR is the performance of a TADOT for Turbine Trip Functions which is performed prior to reactor startup if it has not been performed within the last 31 days. This test shall verify OPERABILITY by actuation of the end devices.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

227

This SR is modified by a Note stating that ~~this Surveillance is not required if it has been performed within verification of the previous 31 days. Trip Setpoint does not have to be performed for this Surveillance. A second Note states that verification of the Trip Setpoint does not have to be performed for this Surveillance.~~ Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical because this test cannot be performed with the reactor at power.

SR 3.3.1.13

This SR is the performance of a COT of the RT's interlocks every 24 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.3.1.13~~
~~REQUIREMENTS~~

~~(continued) This SR ensures the Power Range Neutron Flux Low and the Intermediate Range Neutron Flux trip Functions are not blocked when THERMAL POWER is below the P-10 interlock while in MODE~~

REFERENCES

- ~~1. This Function is derived from a bistable circuit of the Power Range channels. Periodic testing of the P-10 channels is required to verify the setpoint to be less than or equal to the limit. Setpoints must be within the Trip Setpoint of < 6% RTP.~~

~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8). The setpoint shall be left set consistent with the assumptions of the setpoint methodology.~~

~~If the P-10 interlock setpoint is nonconservative, then the Power Range Neutron Flux Low and Intermediate Range Neutron Flux trip Functions are considered inoperable. Alternatively, the P-10 interlock can be placed in the conservative condition (nonblocked). If placed in the nonblocked condition, the SR is met and the Power Range Neutron Flux Low and Intermediate Range Neutron Flux trip Functions would not be considered inoperable.~~

~~The Frequency of 24 months is based on the known reliability of the Functions and multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

SR 3.3.1.14

~~This SR ensures the Source Range Neutron Flux trip Function is not blocked when THERMAL POWER is below the P-6 interlock while in MODE 2. Periodic testing of the P-6 channels is required to verify the setpoint to be less than or equal to the limit.~~

BASES

SURVEILLANCE ~~SR 3.3.1.14 (continued)~~
REQUIREMENTS

~~The difference between the current "as found" values and previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

~~If the P-6 interlock setpoint is nonconservative, then the Source Range Neutron Flux trip Function is considered inoperable. Alternatively, the P-6 interlock can be placed in the conservative condition (nonblocked). If placed in the nonblocked condition, the SR is met and the SRM Function would not be considered inoperable.~~

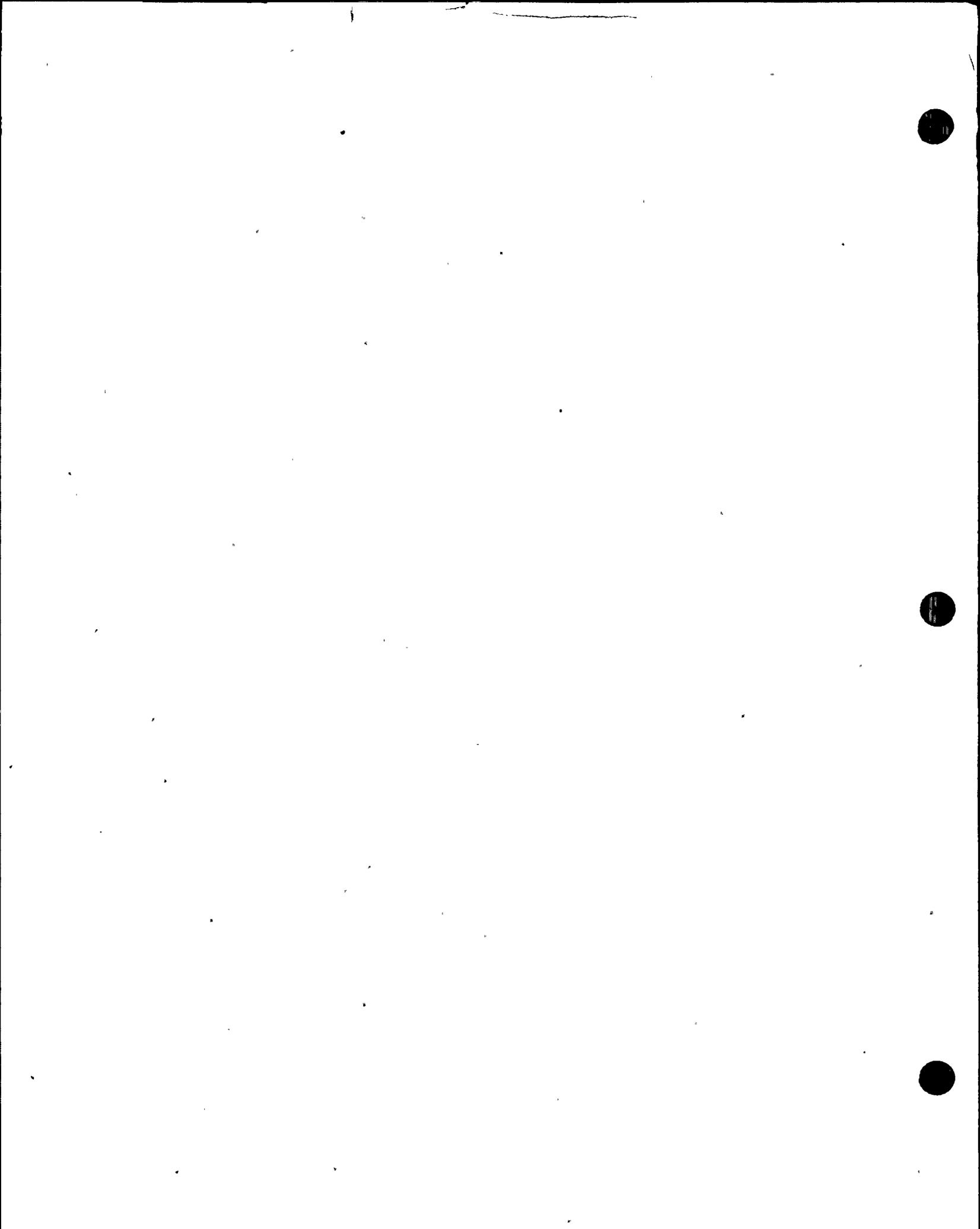
~~The frequency of 24 months is based on the known reliability of the Functions and multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

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SR 3.3.1.15

~~This SR ensures the Pressurizer Pressure Low, Reactor Coolant Flow Low (Two Loops), RCP Breaker Position (Two Loops), and Undervoltage Bus 11A and 11B trip Functions are not blocked when THERMAL POWER is above the P-7 interlock setpoint while in MODE 1. This Function is derived from a bistable circuit indication, > 8.5% RTP as measured by Turbine First Stage Pressure Channels and the Power Range channels. Periodic testing of the P-7 channels is required to verify the setpoint to be less than or equal to the limit.~~

~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~



BASES

~~SURVEILLANCE~~ ~~SR 3.3.1.15~~ (continued)
~~REQUIREMENTS~~

~~If the P 7 interlock setpoint is nonconservative, then the Pressurizer Pressure Low, Reactor Coolant Flow Low (Two Loops), RCP Breaker Position (Two Loops), and Undervoltage Bus 11A and 11B Functions are considered inoperable. Alternatively, the P 7 interlock can be placed in the conservative condition (nonblocked). If placed in the nonblocked condition, the SR is met and the Pressurizer Pressure Low, Reactor Coolant Flow Low (Two Loops), RCP Breaker Position (Two Loops), and Undervoltage Bus 11A and 11B Functions would not be considered inoperable.~~

~~The Frequency of 24 months is based on the known reliability of the Functions and multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

SR 3.3.1.16

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~~This SR ensures the Reactor Coolant Flow Low (Single Loop) and RCP Breaker Position (Single Loop) Functions are not blocked when THERMAL POWER is above the P 8 interlock setpoint while in MODE 1. This Function is derived from a bistable circuit indicating $\geq 50\%$ RTP as measured by the power range channels. Periodic testing of the channels is required to verify the setpoint to be less than or equal to the limit.~~

~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8). the setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

~~If the P 8 interlock setpoint is nonconservative, then the Reactor Coolant Flow low (Single Loop) and RCP Breaker Position (Single Loop) trip Functions are considered inoperable. Alternatively, the P 8 interlock can be placed in the conservative condition (nonblocked). If placed in the nonblocked condition, the SR is met and the Reactor Coolant Flow Low (Single Loop) trip Functions would not be considered inoperable.~~

BASES

~~SURVEILLANCE~~ ~~SR 3.3.1.16~~ (continued)
~~REQUIREMENTS~~

~~The Frequency of 24 months is based on the known reliability of the Functions and multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

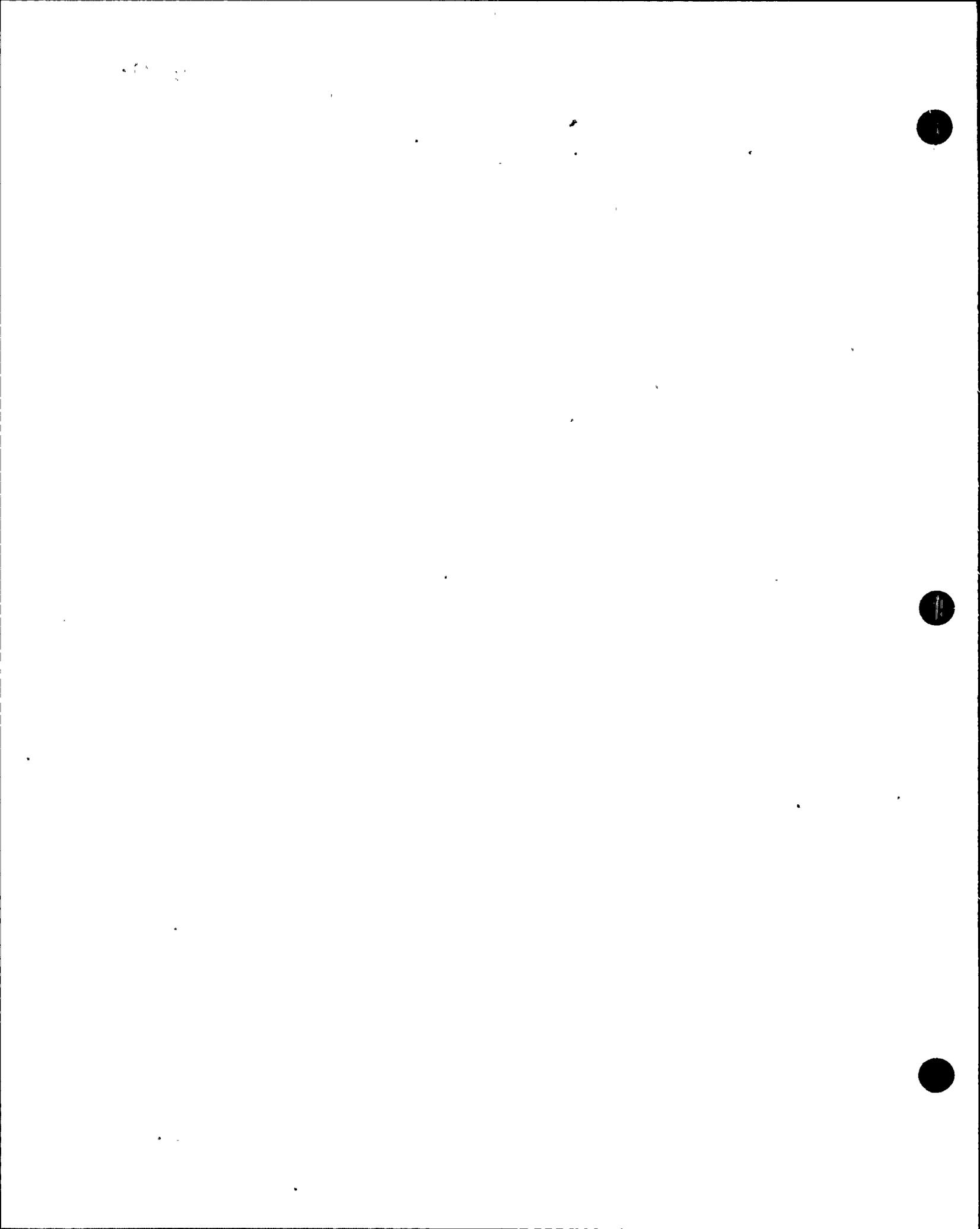
SR 3.3.1.17

~~This SR ensures the Turbine Trip Functions are not blocked when THERMAL POWER is above the P-9 interlock while in MODE 1. This Function is derived from a bistable circuit indicating > 50% RTP as measured by the power range channels. Condenser pressure and circulating water pump breaker status is also an input to P-9 but are not required to be verified by this SR. Periodic testing of the P-9 channels is required to verify the setpoint to be less than or equal to the limit.~~

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~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

~~If the P-9 interlock setpoint is nonconservative, then the Turbine Trip Functions are considered inoperable. Alternatively, the P-9 interlock can be placed in the conservative condition (nonblocked). If placed in the nonblocked condition, the SR is met and the Turbine Trip Functions would not be considered inoperable.~~



BASES

REFERENCES

1. Atomic Industry Forum (AIF) GDC 14, Issued for comment July 10, 1967.
2. 10 CFR 100.
3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
4. UFSAR, Chapter 7.
5. UFSAR, Chapter 6.
6. UFSAR, Chapter 15.
7. IEEE-279-1971.
8. RG&E Engineering Work Request (EWR) 5126, "Guidelines for Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
9. ~~WCAP-14333, May 1995.~~
10. ~~WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.~~

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~~11. [REDACTED]~~

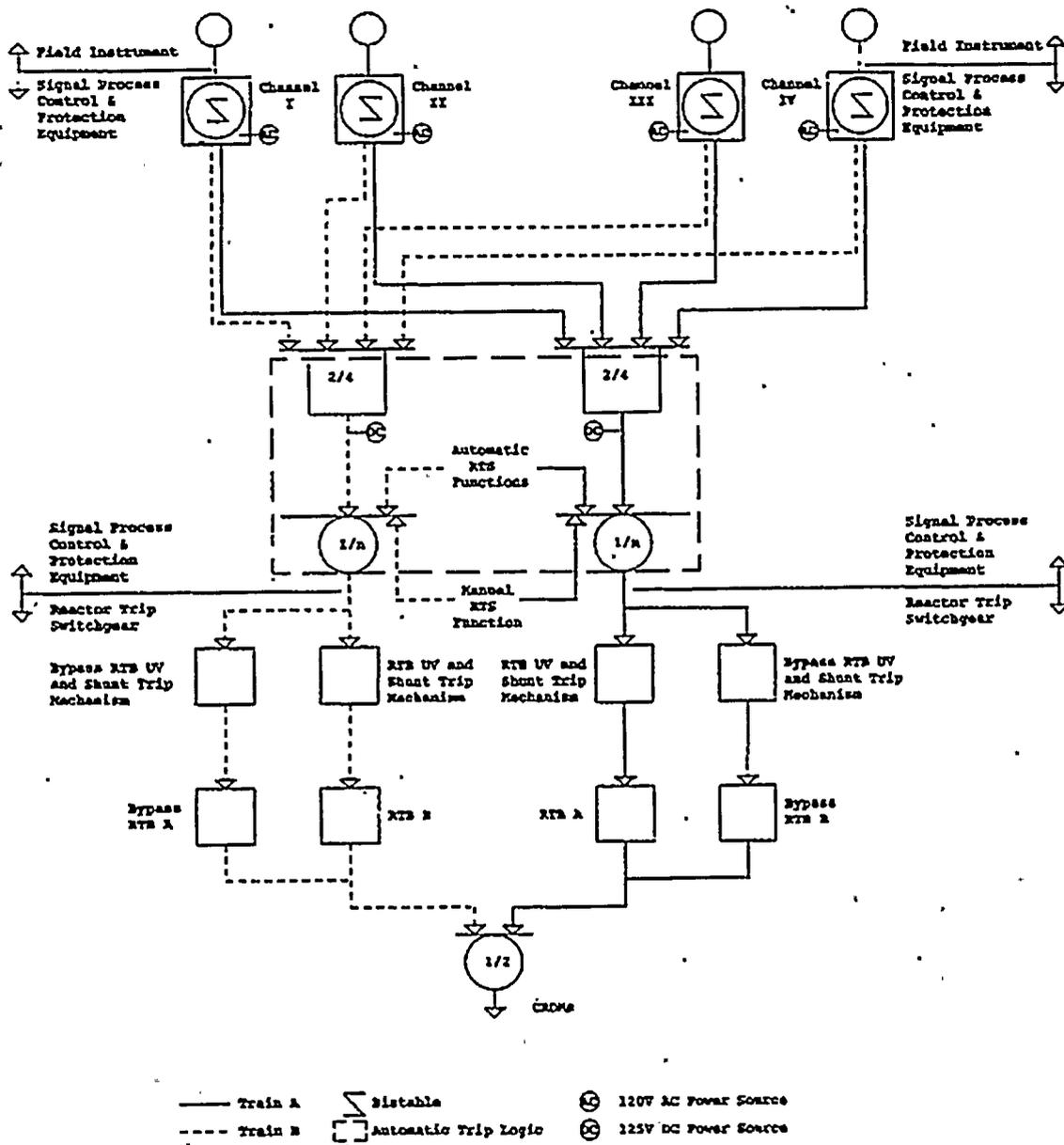


Figure B 3.3.1-1

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND Atomic Industrial Forum (AIF) GDC 15 (Ref. ~~Westinghouse Technical Bulletin, Number NSD TB 92-14-R0, "Instrumentation Calibration at Reduced Power," dated January 18, 1993.~~)

No changes

(continued)

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(continued)



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~~ESFAS Instrumentation
B-3.3.2~~

~~B 3.3 INSTRUMENTATION~~

~~B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation~~

BASES

No changes

~~BACKGROUND~~ — ~~Atomic Industrial Forum (AIF) GDC 15 (Ref. 1)~~ requires that protection systems be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

The ESFAS initiates necessary safety systems, based on the values of selected plant parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into two distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 2):

- Field transmitters or process sensors; and
- Signal processing equipment.

These modules are discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). To account for calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These

(continued)



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statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor.

(continued)



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BACKGROUND
(continued)Signal Processing Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 6 (Ref. 3), Chapter 7 (Ref. 2), and Chapter 15 (Ref. 4). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 5).

The actuation of ESF components is accomplished through master and slave relays. The protection system energizes the master relays appropriate for the condition of the plant. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices.

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Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, ~~Pressurizer-SI-Pressurizer~~ Pressure-Low is a primary actuation signal for small break loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as ~~backups~~ ~~anticipatory~~ ~~actions~~ to Functions that were credited in the accident analysis (Ref. 4).

This LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of three or four channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single failure disables the ESFAS.

(continued)

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The LCO and Applicability of each ESFAS Function are provided in Table 3.3.2-1. Included on Table 3.3.2-1 are Allowable Values and Trip Setpoints for all applicable ESFAS Functions. Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated within the LCOs, including any Required Actions that are in effect at the onset of the DBA and the equipment functions as designed.

The Trip Setpoints are the nominal limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 2, 3, and 4. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.2-1 are therefore conservatively adjusted with respect to the analytical limits (i.e., Allowable Values) used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 6. If the measured setpoint exceeds the Trip Setpoint Value, the bistable is considered OPERABLE unless the Allowable Value as specified in plant procedures is exceeded. The Allowable Value specified in the plant procedures bounds that provided in Table 3.3.2-1 since the values in the table are typically those used in the accident analysis.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 have been confirmed based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

(continued)

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The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protection functions provided in Table 3.3.2-1 are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to $< 2200^{\circ}\text{F}$); and
2. Boration to ensure recovery and maintenance of SDM ($k_{\text{eff}} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Containment Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;
- Feedwater Isolation; and
- Start of motor driven auxiliary feedwater (AFW) pumps.

(continued)



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1. Safety Injection (continued)

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses; and
- Start of AFW to ensure secondary side cooling capability.

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a. ~~Safety Injection~~ Manual Injection = Manual Initiation

This LCO requires one channel per train to be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. The operator can initiate SI at any time by using either of two pushbuttons on the main control board. This action will cause actuation of all components with the exception of Containment Isolation and Containment Ventilation Isolation.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one pushbutton and the interconnecting wiring to the actuation logic cabinet. Each pushbutton actuates both trains. This configuration does not allow testing at power.

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a. ~~Safety Injection - Manual Injection - Manual~~
Initiation (continued)

This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

~~Also, this Function is not required in MODE 4 since it does not actuate Containment Isolation or Containment Ventilation Isolation.~~

~~Safety Injection - Automatic Actuation Logic and Actuation Relays~~

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MODES 1, 2, and 3b.

~~This LCO requires two trains to be OPERABLE in~~
~~Safety Injection - Automatic Actuation Logic and Actuation Relays~~

~~This LCO requires two trains to be OPERABLE in MODES 1, 2, 3, and 4. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.~~

This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise

(continued)



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prevented from actuating to prevent inadvertent overpressurization of plant systems.

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c. ~~Safety Injection Containment~~
~~Pressure-High Injection Containment Pressure-High~~

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters and electronics are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure-High must be OPERABLE in MODES 1, 2, and 3, and 4 because there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, Containment Pressure-High is not required to be OPERABLE because there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. ~~Safety Injection Pressurizer~~
~~Pressure-Low Injection Pressurizer Pressure-Low~~

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) atmospheric relief or safety valve;

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- SLB;

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d. ~~Safety Injection-Pressurizer~~

~~Pressure-LowInjection-Pressurizer Pressure-Low~~
(continued)

- Rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Since there are dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the interlock setpoint. Automatic SI actuation below this interlock setpoint is performed by the Containment Pressure-High signal.

This Function is not required to be OPERABLE in MODE 3 below the Pressurizer Pressure interlock setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

(continued)

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e. ~~Safety Injection- Steam Injection - Steam Line~~
~~Pressure - Low Pressure - Low~~

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG atmospheric relief or an SG safety valve.

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Steam line pressure transmitters provide control input, but the control function cannot initiate events that the Function acts to mitigate. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line.

~~Each steam line is considered a separate function for the purpose of this LCO.~~

With the transmitters located in the Intermediate Building, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

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- e. ~~Safety Injection- Steam Injection- Steam Line~~
~~Pressure-Low-Pressure-Low~~
(continued)

Steam Line Pressure-Low must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) when a secondary side break or stuck open SG atmospheric relief or safety valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the interlock setpoint. Below the interlock setpoint, a feed line break is not a concern. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the plant to cause an accident.

2. Containment Spray (CS)

CS provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and
3. Adjusts the pH of the water in containment sump B after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

(continued)



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2. CS (continued)

CS is actuated manually or by Containment Pressure-High High. The CS actuation signal starts the CS pumps and aligns the discharge of the pumps to the CS nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the CS pumps and mixed with a sodium hydroxide solution from the spray additive tank. During the recirculation phase of accident recovery, the spray pump suctions are manually shifted to containment sump B if continued CS is required.

a. ~~CS-Manual~~ CS-Manual Initiation

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The operator can initiate CS at any time from the control room by simultaneously depressing two CS actuation pushbuttons. Because an inadvertent actuation of CS could have serious consequences, two pushbuttons must be simultaneously depressed to initiate both trains of CS.

~~Therefore, the inoperability of either pushbutton fails both trains of manual initiation.~~

Manual initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

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(continued)

b. ~~CS-Automatic~~ ~~CS-Automatic~~ Actuation Logic and Actuate Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

c. ~~CS-Containment~~ ~~CS-Containment~~ Pressure-High High

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This signal provides protection against a LOCA or an SLB inside containment. The transmitters are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is the only ~~ESFAS~~ Function that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate CS, since the consequences

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of an inadvertent actuation of CS could be serious.

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c. ~~CS-Containment~~ ~~CS-Containment~~ Pressure-High High
(continued)

The Containment Pressure-High High instrument function consists of two sets with three channels in each set. Each set is a two-out-of-three logic where the outputs are combined so that both sets tripped initiates CS. Each set is considered a separate function for the purposes of this LCO. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure-High High must be OPERABLE in MODES 1, 2, 3 and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and selected process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a LOCA.

(continued)



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3. Containment Isolation (continued)

Containment Isolation signals isolate all automatically isolatable process lines, except feedwater lines, main steam lines, and component cooling water (CCW). The main feedwater and steam lines are isolated by other functions since forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW enhances plant safety by allowing operators to use forced RCS circulation to cool the plant. Isolating CCW may ~~forerequire~~ the use of feed and bleed cooling, which could prove more difficult to control.

a. Containment Isolation Manual
Isolation Isolation Manual Initiation

Manual Containment Isolation is actuated by either of two pushbuttons on the main control board. Either pushbutton actuates both trains. Manual initiation of Containment Isolation also actuates Containment Ventilation Isolation.

Manual initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate ~~unit~~ ~~plant~~ conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

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b. Containment
~~Isolation-Automatic Isolation-Automatic~~ Actuation
Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate ~~unit-plant~~ conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

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c. ~~Containment Isolation-Safety Isolation-Safety~~
Injection

Containment Isolation is also initiated by all Functions that automatically initiate SI. The Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating Functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Closure of the main steam isolation valves (MSIVs) and ~~their associated~~ non-return check valves limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures

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(continued)

BASES

a source of steam for the turbine driven AFW pump
during a feed line break.

(continued)

BASES

APPLICABLE

SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

(167)

a. ~~Steam Line Isolation - Manual Isolation - Manual~~
Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two actuation devices (one pushbutton and one switch) on the main control board for each MSIV. Each device can initiate action to immediately close its respective MSIV. The LCO requires one channel (device) per loop to be OPERABLE. ~~Each loop is not considered a separate function since there is only one required per loop.~~

Manual initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

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b. Steam Line
~~Isolation - Automatic Isolation - Automatic~~
Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

(continued)

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APPLICABLE

(167)

SAFETY ANALYSES,
LCO, and
APPLICABILITY

- b. Steam Line
~~Isolation-Automatic Isolation-Automatic~~ Actuation
Logic
and Actuation Relays (continued)

Automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

- c. Steam Line Isolation-Containment
~~Pressure-High Isolation-Containment Pressure-High~~ High

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This Function actuates closure of both MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters are located outside containment with the sensing lines passing ~~through~~ containment penetrations to sense the containment atmosphere in three different locations. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties. Containment Pressure-High High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic.

(continued)



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APPLICABLE

SAFETY ANALYSES,
LCO, and
APPLICABILITY

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- c. Steam Line
~~Isolation-Containment Isolation-Containment~~
~~Pressure-High Pressure-High High~~ (continued)

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, because there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The steam line isolation function must be OPERABLE in MODES 2 and 3, unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6 the steam line isolation function is not required to be OPERABLE because there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

- d. Steam Line Isolation-High Isolation-High Steam
Flow Coincident
With Safety Injection and Coincident With
~~F_{avg} - Low I_{avg} - LOW~~

(167)

This function provides closure of the MSIVs during an SLB or inadvertent opening of an SG atmospheric relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

(169)

Two steam line flow channels per steam line are required to be OPERABLE for this function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. ~~Each steam line is considered a separate function for the purpose of this LCO.~~ The steam flow transmitters provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues. The one-out-of-two configuration allows online

(continued)



BASES

testing because trip of one high steam flow
channel is not sufficient to cause initiation.

(continued)

BASES

APPLICABLE

SAFETY ANALYSES,
LCO, and
APPLICABILITY

(169)

- d. ~~Steam Line Isolation-High Isolation-High Steam~~
~~Flow Coincident~~
~~With Safety Injection and Coincident With~~
 ~~T_{avg} -Low T_{avg} -Low~~ (continued)

With the transmitters (d/p cells) located inside containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

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Two channels of T_{avg} per loop are required to be OPERABLE for this Function. Each loop is considered a separate Function for the purpose of this LCO. The T_{avg} channels are combined in a logic such that any two of the four T_{avg} channels tripped in conjunction with SI and one of the two high steam line flow channels tripped causes isolation of the steam line associated with the tripped steam line flow channels. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels per loop in a two-out-of-four configuration ensures no single failure disables the T_{avg} -Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore; additional channels are not required to address control protection interaction issues.

(continued)

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APPLICABLE (161)
SAFETY ANALYSES,
LCO, and
APPLICABILITY

- d. ~~Steam Line Isolation-High Isolation-High Steam~~
~~Flow Coincident-~~
~~With Safety Injection and Coincident With~~
~~T_{avg}-Low T_{avg}-Low~~ (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

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- e. ~~Steam Line Isolation-High Isolation-High High~~
~~Steam Flow~~
~~Coincident With Safety Injection~~

This Function provides closure of the MSIVs during a steam line break (or inadvertent opening of an SG atmospheric relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

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Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high-high steam flow in one steam line. ~~Each steam line is considered a separate function for the purpose of this LCO.~~ The steam flow transmitters provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

(continued)

BASES

APPLICABLE (67)
SAFETY ANALYSES,
LCO, and
APPLICABILITY

- e. Steam Line Isolation-High Isolation-High High Steam Flow Coincident With Safety Injection (continued)

The main steam lines isolate only if the high-high steam flow signal occurs coincident with an SI signal. Steamline isolation occurs only for the steam line associated with the tripped steam flow channels. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIV's are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

5. Feedwater Isolation

The primary function of the Feedwater Isolation signals is to prevent and mitigate the effects of high water level in the SGs which could cause carryover of water into the steam lines and result in excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

This Function is actuated by either a SG Water Level-High or by an SI signal. The Function provides feedwater isolation by closing the ~~MFRVs~~ ~~Main Feedwater Regulating Valves (MFRVs)~~ and the associated bypass valves. In addition, on an SI signal, the AFW System is automatically started, and the MFW pump breakers are opened which closes the MFW pump discharge valves. The SI signal was discussed previously.

a. ~~Feedwater Isolation-Automatic Isolation-Automatic Actuation~~
Logic and Actuation Relays

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Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all ~~Main Feedwater Regulating Valves~~ ~~MFRVs~~ and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)

BASES

APPLICABLE

SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

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169

169

b. ~~Feedwater Isolation- Steam Isolation- Steam~~
~~Generator Water~~
~~Level- High Level- High~~

The Steam Generator Water Level - High Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all Main Feedwater Regulating Valves MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - High is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

c. ~~Feedwater Isolation- Safety Isolation- Safety~~
~~Injection~~

The Safety Injection Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all Main Feedwater Regulating Valves MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

(continued)



BASES

APPLICABLE c. ~~Feedwater Isolation-Safety Isolation-Safety Injection~~

SAFETY ANALYSES, ~~_____~~ (continued)

LCO, and

~~Feedwater Isolation is also initiated by all~~ APPLICABILITY

(169)

~~Feedwater Isolation is also initiated by all~~ Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The preferred system has two motor driven pumps and a turbine driven pump, making it available during normal plant operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break (depending on break location). A Standby AFW (SAFW) is also available in the event the preferred system is unavailable. The normal source of water for the AFW System is the condensate storage tank (CST) which is not safety related. Upon a low level in the CST the operators can manually realign the pump suctions to the Service Water (SW) System which is the safety related water source. The AFWSW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately also is the safety related water source for the SAFW System.

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The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately while the SAFW System is only manually initiated and aligned.

(continued)

BASES

~~Auxiliary Feedwater Automatic Actuation Logic
and Actuation Relays~~

(16)

~~Actuation logic consists of all circuitry housed
within the actuation subsystems, including the
initiating relay contacts responsible for
actuating the ESF equipment.~~

(continued)

BASES

~~APPLICABLE~~ ~~auxiliary Feedwater Manual Initiation~~

control switches

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~~The operator can initiate AFW or SAFW at any time by using pushbuttons on the Main Control board (one pushbutton for each pump in each system).~~

~~SAFETY ANALYSES, LCO, and~~ ~~APPLICABILITY~~ ~~Auxiliary Feedwater Automatic Actuation Logic and Actuation Relays (continued)~~

~~Automatic initiation~~ ~~This action will cause actuation of Auxiliary Feedwater must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor their respective pumps and valves.~~

(continued)

BASES

SURVEILLANCE SR 3.3.6.4
REQUIREMENTS

(continued)

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations are tested for the CREATS actuation instrumentation. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is acceptable based on instrument reliability and operating experience.

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REFERENCES

1. UFSAR, Section 6.4.
-

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~~In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.~~ APPLICABLE a. ~~This Auxiliary Feedwater Manual~~

Initiation

SAFETY ANALYSES, (continued):

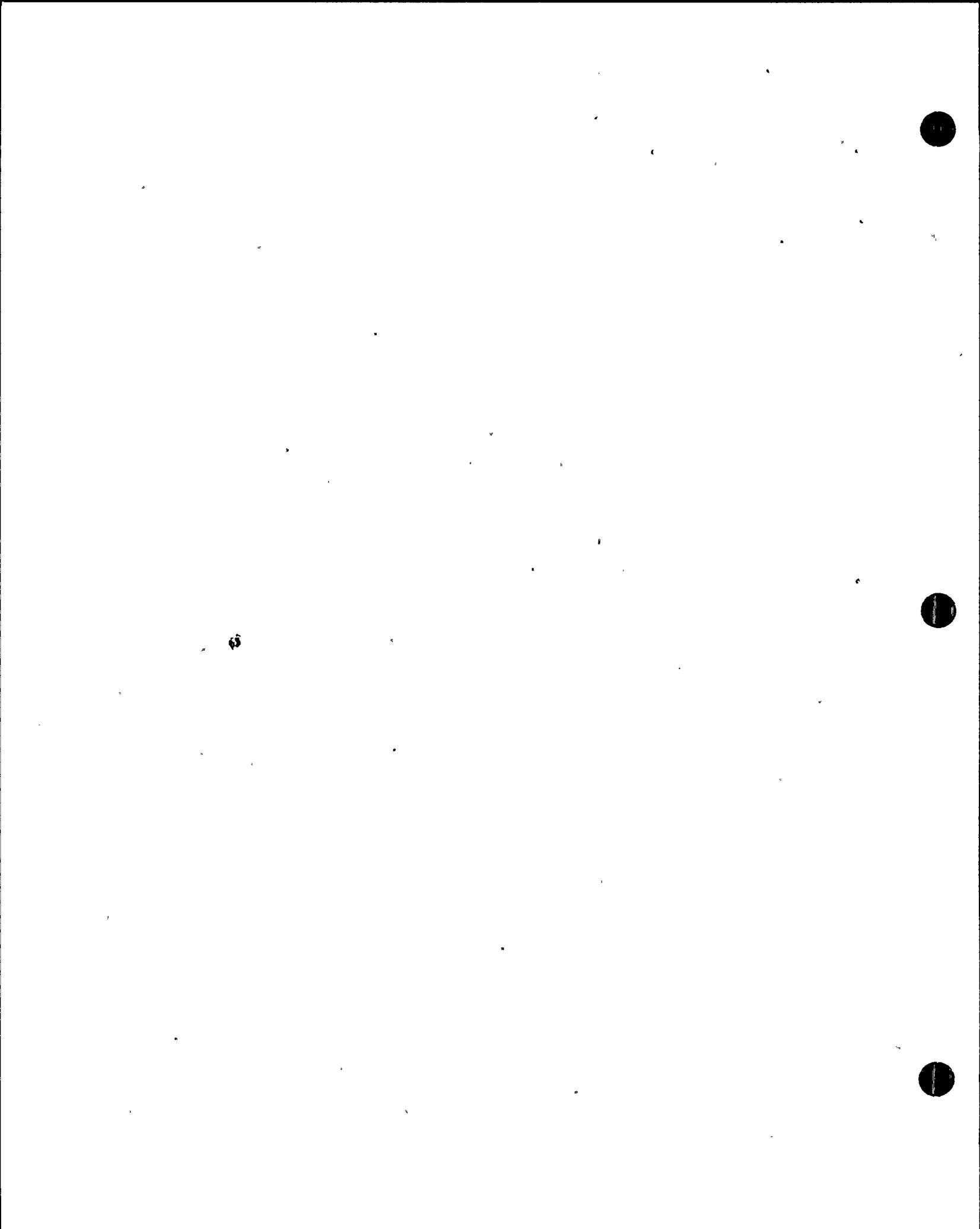
LCO, and

APPLICABILITY

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The LCO for the Manual Initiation Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated inensures the reactor to require the SGs as a heat sinkproper amount of redundancy is maintained to ensure the operator has manual AFW and SAFW initiation capability.—

(continued)



BASES

APPLICABLE

~~remain the heat sink for the reactor.~~ ~~Auxiliary Feedwater Steam Generator Water~~

~~SAFETY ANALYSES, Level Low Low~~
~~LCO, and~~
~~APPLICABILITY~~

~~(continued)~~

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~~bThe LCO requires one channel per pump in each system to be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs~~
~~actuation is not required to be OPERABLE in MODES 1, 2, and 3 to provide protection against a loss of heat sink because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.~~
~~This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.~~

~~b~~ Auxiliary Feedwater Automatic Actuation Logic and Actuation Relays

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~~Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.~~

(continued)



BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY b. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (continued)

Automatic initiation of Auxiliary Feedwater must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

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c. Auxiliary Feedwater - Steam Generator Water Level - Low Low

SG Water Level - Low Low must be OPERABLE in MODES 1, 2, and 3 to provide protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level - Low Low in either SG will cause both motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level - Low Low in both SGs will cause the turbine driven pump to start. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

(continued)

BASES

APPLICABLE
c. Auxiliary Feedwater - Steam Generator Water
Level - Low/Low (continued)
SAFETY ANALYSES, LCO, and
APPLICABILITY

169

Each SG is considered a separate function for the purpose of this LCO. The Allowable Value for SG Water Level - Low Low is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

(continued)

BASES

APPLICABLE ~~ed.~~ ~~Auxiliary Feedwater Safety Feedwater Safety~~
Injection
~~SAFETY ANALYSES,~~

(169)

LCO, and

APPLICABILITY

(continued)

The SI function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink

for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

(continued)

BASES

~~APPLICABLE~~ e.

Auxiliary

~~Feedwater Undervoltage Bus Feedwater Undervoltage Bus 11A and 11B~~

LCO, and
APPLICABILITY
(continued)

SAFETY ANALYSES

The Undervoltage - Bus 11A and 11B Function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

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APPLICABLE

~~A loss of power to 4160 V Bus 11A and 11B will~~

~~be accompanied by a loss of power to both MFW pumps and the subsequent need for some method of~~

~~decay heat removal.~~

~~Auxiliary Feedwater Undervoltage Bus 11A and 11B (continued)~~

~~SAFETY ANALYSES,
LCO, and
APPLICABILITY~~

~~A loss of power to 4160 V Bus 11A and 11B will be accompanied by a loss of power to both MFW pumps and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each bus. Loss of power to both buses will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.~~

~~purpose of this LCO.~~

~~Each bus is considered a separate Function for the~~

Auxiliary Feedwater Trip Of Both Main Feedwater Pumps

~~A Trip of both MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal.~~

(continued)

BASES

Auxiliary Feedwater - Trip Of Both Main
Feedwater

Pumps

(14)

A Trip of both MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal. The MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per AFWMFW pump satisfy redundancy requirements with two-out-of-two logic. Each MFW pump is considered a Separate Function for the purpose of this LCO. A trip of both MFW pumps starts both motor driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.

(continued)

BASES

~~This Function must be OPERABLE in MODES 1 and 2 APPLICABLE~~

f. ~~Auxiliary Feedwater - Trip Of Both Main Feedwater Pumps (continued)~~

SAFETY ANALYSES,
LCO, and
APPLICABILITY

~~This Function must be OPERABLE in MODES 1 and 2~~

(169)

This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, 5, and 6 the MFW pumps are not in operation, and thus pump trip is not indicative of a condition requiring automatic AFW initiation.

ACTIONS — The

~~_____~~
~~_____~~
~~_____~~

ACTIONS

(218)

~~for each inoperable ESFAS Function are identified by the Condition column. A Note has been added in the ACTIONS to clarify the application of Table 3.3.2-1 Completion Time rules. A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.~~

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. As shown on Figure B 3.3.2-1, the ESFAS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Actuation Logic. Therefore, a channel may be inoperable due to the failure of a field instrument, loss of 120 VAC instrument bus power or a bistable failure which affects one or both ESFAS trains. The only exception to this are the Manual ESFAS and Automatic Actuation Logic Functions which are defined strictly on a train basis. The Automatic Actuation Logic consists of all circuitry housed within the actuation subsystem, including the master relays,

(continued)

BASES

slave relays, and initiating relay contacts responsible for activating the ESF equipment.

(continued)

BASES

A.1

~~Condition A applies to all ESFAS protection functions.~~

~~Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable. ACTIONS~~

~~A.1~~

~~(continued)~~

~~Condition A applies to all ESFAS protection functions.~~

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~~Condition A addresses the situation where one channel or train for one or more Functions are inoperable. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.~~

(continued)

D
BASES

ACTIONS ~~A.1~~ (continued)

~~When the number of inoperable channels in an ESFAS Function exceed those specified in all related Conditions associated with an ESFAS Function, then the plant is outside the safety analysis.~~

(218) When the number of inoperable channels in an ESFAS Function exceed those specified in all related Conditions associated with an ESFAS Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if the ESFAS function is applicable in the current MODE of operation.

B.1

Condition ~~B~~ applies to the channel or train orientation of the ESFAS for the following Functions:

- ~~• Manual Initiation of SI;~~
- ~~• Manual Initiation of CS;~~
- ~~• Manual Initiation of Containment Isolation;~~
- ~~• Manual Initiation of Steam Line Isolation;~~
- ~~• Undervoltage Bus 11A and 11B; and~~
- ~~• Trip ~~AFW~~ Trip of Both MFW Pumps, pumps.~~ ^{ESFAS Function}

(218) ~~If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. For the Manual Initiation Functions, the specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each the nature of this Function (except for CS); the available redundancy, and the low probability of an event occurring during this interval.~~

(continued)

BASES

~~For the Undervoltage-Bus 11A and 11B Function and Trip of Both MFW Pumps Functions, the specified Completion Time of 48 hours is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. The~~

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If the Required Action and Completion Time of 48 hours for Trip of Both MFW Pumps Function Condition B is consistent with Reference 7 not met, the plant must be brought to a MODE in which the LCO does not apply.

(continued)

BASES

~~ACTIONS~~ C.1
~~(continued)~~

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~~If the channel for Function 6.e cannot be restored to OPERABLE status within the required Completion Time of Condition B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions~~ **MODE 3** ~~from full power conditions in an orderly manner and without challenging plant systems.~~

(continued)

BASES

ACTIONS
(continued)

~~D.1 and~~

~~Condition D applies to the following ESFAS Functions:~~

- ~~• Manual Initiation of SI;~~
- ~~• Manual Initiation of Steam Line Isolation; and~~
- ~~• AFW - SG Water Level - Low Low; and~~
- ~~• Manual Initiation of AFW;~~
- ~~• AFW - Undervoltage - Bus 11A and 11B.~~

additional AFW
actuation channels
available besides the
SG water level - low low
and Undervoltage - Bus
11A and 11B AFW
Initiation Functions,

~~If the channel for Function 6 is inoperable, 48 hours is allowed to restore it to OPERABLE status. d or the train. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each - manual initiation Function and the low probability of an event occurring during this interval.~~



~~a or 4E.a cannot be restored to OPERABLE status within the required Completion Time of 1~~

~~Condition B, the plant must be brought to a MODE in which the LCO does not apply. E applies to the automatic actuation logic and actuation relays for the following ESFAS functions:~~

- ~~• Steam Line Isolation;~~
- ~~• Feedwater Isolation; and~~
- ~~• AFW.—~~

~~To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. Condition E addresses the train orientation of the protection system and the master and slave relays. The allowed if one train is inoperable, a Completion times are reasonable, based on operating experience, to reach Time of 6 hours is allowed to restore the required plant conditions from full power conditions in an orderly manner and without challenging plant systems train to OPERABLE status.~~

(continued)

BASES

(218)

This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this time interval. The Completion Time of 6 hours is consistent with Reference 7.

(continued)

BASES

2

~~If the train for Function 2~~ ACTIONS
~~1, a or 3~~

(continued)

Condition F applies to the following Functions:

- Steam Line Isolation - Containment Pressure - High High;
- Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low;
- Steam Line Isolation - High - High Steam Flow Coincident With Safety Injection; and
- Feedwater Isolation - SG Water Level - High;
- AFW - SG Water Level - Low Low; and
- AFW - Undervoltage - Bus - 11A and 11B.

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~~a cannot be restored to OPERABLE status within the required Completion Time of Condition B, the plant must be brought to a MODE in which the LCO does not apply. Applies to Functions that typically operate on two out of three logic. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

(continued)

BASES

~~ACTIONS~~ ~~F.1~~
~~(continued)~~

~~Condition F applies to the automatic actuation logic and actuation relays for the following Functions:~~

- ~~• SI;~~
- ~~• CS;~~
- ~~• Containment Isolation;~~
- ~~• Steam Line Isolation;~~
- ~~• Feedwater Isolation; and~~
- ~~• Auxiliary Feedwater.~~

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~~Condition F addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 24 hours is allowed to restore the train to OPERABLE status. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.~~

~~If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.~~

~~The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. This 4 hours applies to each of the remaining OPERABLE channels.~~

~~The Completion Time of 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 7.~~

(continued)

BASES

ACTIONS **G.1**
(continued)

If the Required Actions and Completion Times of Conditions D, E, or F are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

Condition H applies to the following ESFAS functions:

- Manual Initiation of CS; and
- Manual Initiation of Containment Isolation.

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If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each Function (except for CS) and the low probability of an event occurring during this interval.

I.1

Condition I applies to the automatic actuation logic and actuation relays for the following Functions:

- SI;
- CS; and
- Containment Isolation.

(continued)

BASES

ACTIONS I.1 (continued)

Condition I addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. ~~The Completion Time of 24 hours is consistent with Reference 8.~~

G.1 and G.2

~~If the train for Function 1.b, 4.b, 5.a, or 6.a cannot be restored to OPERABLE status within the required Completion Time of Condition F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The Completion Time of 6 hours is consistent with Reference 7.~~

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J.1

Condition J applies to the following functions:

- SI - Containment Pressure - High; and
- CS - Containment Pressure - High High

Condition J applies to functions that operate on a two-out-of-three logic (for CS - Containment Pressure - High High there are two sets of this logic). Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

(continued)



BASES

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

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The Completion time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

(continued)

BASES

ACTIONS K-1
(continued)

(218)

If the Required Actions and Completion Times of Conditions H, I, or J are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

~~ACTIONS~~ ~~HL.1~~

~~Condition 1 applies to the following Functions:~~

- ~~• SI = Pressurizer Pressure = Low; and H~~
- ~~• SI = Steam Line Pressure = Low.~~

~~2~~

~~(continued)~~

~~If the train for Function 2 Condition 1 applies to Functions that operate on a two-out-of-three logic. b or 3 Therefore, failure of one channel places the Function in a two-out-of-two configuration. b cannot be restored to OPERABLE status within the required Completion Time of Condition F, the plant One channel must be brought tripped to a MODE place the Function in which the LCO does not apply a one-out-of-two configuration that satisfies redundancy requirements.~~

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(218)
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~~To achieve this status If one channel is inoperable, the plant must be brought to at least MODE 3 within a Completion Time of 6 hours and to MODE 5 within 36 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions Placing the channel in an orderly manner the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and without challenging plant systems allows operation to continue.~~

~~The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels.~~

~~Condition I The 4 hours applies to each of the following Functions:~~

- ~~• Containment Pressure High;~~
- ~~• Containment Pressure High High;~~
- ~~• Containment Pressure High High;~~

(continued)

BASES

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218
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- ◆ ~~Pressurizer Pressure Low;~~
- ◆ ~~Steam Line Pressure Low;~~
- ◆ ~~High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} Low;~~
- ◆ ~~High High Steam Flow Coincident With Safety Injection;~~
- ◆ ~~SG Water Level High; and~~
- ◆ ~~SG Water Level Low Low remaining OPERABLE channels.-~~

~~Condition I applies to Functions that typically operate on two out of three logic. Therefore, failure of one channel places the Function in a two out of two configuration. One channel must be tripped to place the Function in a one out of two configuration that satisfies redundancy requirements.~~

(continued)

BASES

ACTIONS ~~I.1~~ (continued)

~~If one channel is inoperable, a Completion Time of 72 hours is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.~~

~~The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The Completion Time of 72 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 12 hours allowed for testing, are justified in Reference 8.~~

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213
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J.1 and J.2

~~If the channel for Function 1.d or 1.e cannot be restored to OPERABLE status within the required Completion Time of Condition 1, the plant must be brought to a MODE in which the LCO does not apply. The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.~~

(continued)

BASES

ACTIONS M.1
(continued)

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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218
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K.1 and K.2

~~If the channel for Function 1.c, 4.c, 4.d, 4.e, 5.b, or 6.b cannot be restored to OPERABLE status within the required Completion Time of Condition I, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

(continued)

BASES

~~ACTIONS~~ L.1 and L.2
~~(continued)~~

218

~~If the channel for Function 2.c cannot be restored to OPERABLE status within the required Completion Time of Condition I, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, and without challenging plant systems.~~

SURVEILLANCE
REQUIREMENTS

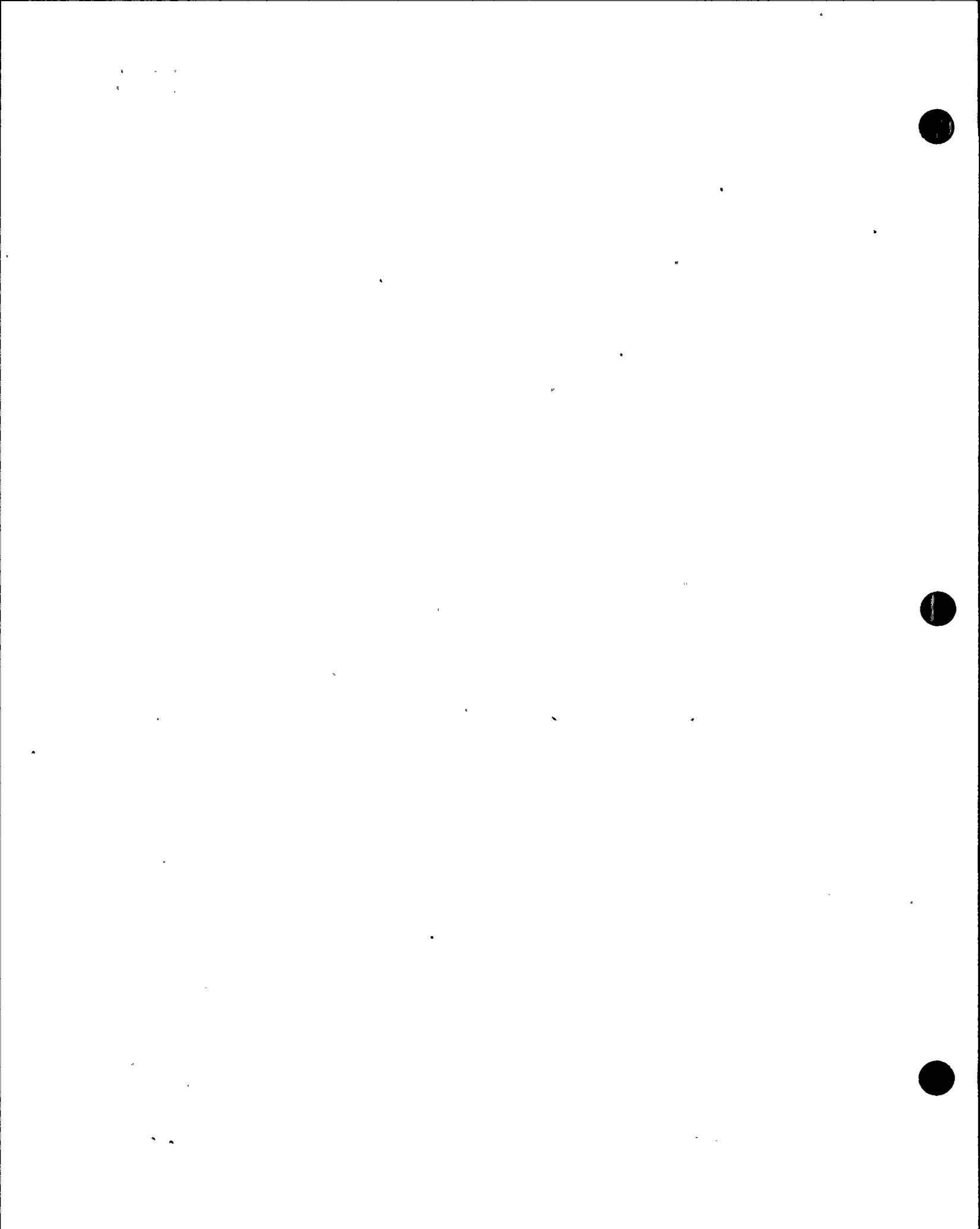
The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

Note 1 has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

N.1

Condition N applies if an AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or S-AFW pump must be declared inoperable and the applicable conditions of LCO 3.7.5 "Auxiliary Feedwater (AFW) System" must be immediately entered. Each AFW manual initiation switch controls one AFW or S-AFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

(continued)



BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

(169)

Note 2 has been added to indicate that, when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 12 hours, provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 12 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the assumption that 12 hours is the average time required to perform channel surveillance. Based on engineering judgement, 12 hour testing allowance does not significantly reduce the probability that the ESFAS instrumentation will trip when necessary.

SURVEILLANCE — SR 3.3.2.1

REQUIREMENTS
(continued)

Performance of the CHANNEL CHECK once every 12 hours ensures that ~~instrumentation has not occurred~~ CHANNEL CALIBRATION for the following ESFAS Functions: CHECK

(169)

- SI - Containment Pressure - High;
- SI - Pressurizer Pressure - Low;
- SI - Steam Line Pressure - Low;
- CS - Containment Pressure - High High;
- Steam Line Isolation - Containment Pressure - High High;
- Steam Line Isolation - High Steam Flow Coincident with SI and T_{avg} - Low;
- Steam Line Isolation - High - High Steam Flow Coincident with SI;
- Feedwater Isolation - SG Water Level - High; and
- AFW - SG Water Level - Low Low.

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**

SR 3.3.2.1 (continued)

(169)

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

CHANNEL CHECK acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 12 hours is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

This SR is the performance of a COT every 92 days for the following ESFAS functions:

(169)

- SI - Containment Pressure - High;
- SI - Pressurizer Pressure - Low;
- SI - Steam Line Pressure - Low;
- CS - Containment Pressure - High High;
- Steam Line Isolation - Containment Pressure - High High;

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.2 (continued)

- * Steam Line Isolation = High Steam Flow Coincident with SI and T_{avg} = Low;
- * Steam Line Isolation = High = High Steam Flow Coincident with SI;
- * Feedwater Isolation = SG Water Level = High; and
- * AFW = SG Water Level = Low Low; and

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AFW - Undervoltage - Bus 11A and 11B.
 A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found to be within the Allowable Values specified in Table 3.3.1-13.3.2-1 and established plant procedures. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 92 days is consistent with in Reference 7. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SURVEILLANCE — SR 3.3.2.3

REQUIREMENTS
(continued)

218

This SR is the performance of a TADOT every 92 days. This test is a check of the Undervoltage - Bus AFW - Undervoltage - Bus 11A and 11B Function.

The test includes trip devices that provide actuation signals directly to the protection system. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency of 92 days is adequate based on industry operating experience, considering instrument reliability and operating history data.

(continued)

BASES

(continued)

BASES

SURVEILLANCE

SR 3.3.2.4

(continued)

REQUIREMENTS

This SR is the performance of a TADOT every 24 months. This test is a check of the SI, CS, Containment Isolation, Steam Line Isolation, and AFW Manual Initiations, Automatic Actuation Logic, and Trip and the AFW-Trip of Both MFW Pumps Functions. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock logic state. Each function is tested up to, and the verification of the required logic output including, the master transfer relay coils. Relay and contact operation is verified by the actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Manual Initiation, Automatic Actuation Logic Initiations, and Trip AFW-Trip of Both MFW Pumps Functions have no associated setpoints.

218

SR 3.3.2.5

This SR is the performance of a CHANNEL CALIBRATION every 24 months of the following ESFAS Functions:

- SI - Containment Pressure - High;
- SI - Pressurizer Pressure - Low;
- SI - Steam Line Pressure - Low;
- CS - Containment Pressure - High High;
- Steam Line Isolation - Containment Pressure - High High;
- Steam Line Isolation - High Steam Flow Coincident with SI and T_{avg} - Low;
- Steam Line Isolation - High - High Steam Flow Coincident with SI;
- Feedwater Isolation - SG Water Level - High; and

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(continued)

BASES

AFW-SG Water Level-Low Low.—

169

~~CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor.~~ AFW-Undervoltage-Bus 11A and 11B.—

(continued)

11 11 11

11 11 11

11 11 11



BASES

~~The test verifies that the channel responds to~~ SURVEILLANCE

SR 3.3.2.5 (continued)

REQUIREMENTS

(169)

CHANNEL CALIBRATION is a measured parameter with complete check of the necessary range and accuracy instrument loop including the sensor.

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.3.2.5~~ (continued)
~~REQUIREMENTS~~

~~CHANNEL CALIBRATIONS must be performed consistent with the test verifies that the channel responds to a measured parameter within the assumptions of the plant specific setpoint methodology necessary range and accuracy.~~

169

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the plant specific setpoint methodology. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.2.6

169

This SR ensures the Pressurizer ~~SI - Pressurizer~~ Pressure - Low and Steam ~~SI - Steam~~ Line Pressure - Low Functions are not bypassed when pressurizer pressure > 2000 psig while in MODES 1, 2, and 3. Periodic testing of the pressurizer pressure channels is required to verify the setpoint to be less than or equal to the limit.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 6). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

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If the pressurizer pressure interlock setpoint is nonconservative, then the Pressurizer Pressure - Low and Steam Line Pressure - Low Functions are considered inoperable. Alternatively, the pressurizer pressure interlock can be placed in the conservative condition ~~(nonbypass)~~ (nonbypassed). If placed in the nonbypassed condition, the SR is met and the Pressurizer Pressure - Low and Steam Line Pressure - Low Functions would not be considered inoperable.

(continued)

BASES

(continued)

BASES

REFERENCES — 1 SURVEILLANCE SR 3.3.2.7

REQUIREMENTS

(continued) This SR is the performance of an ACTUATION LOGIC TEST on all ESFAS Automatic Actuation Logic and Actuation Relays

Functions every 24 months. — Atomic Industrial Forum (AIF) GDC 15, Issues for Comment July 10, 1967. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock state and verification of the required logic output. Relay and contact operation is verified by a continuance check or actuation of the end device.

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The Frequency of 24 months is based on operating experience and the need to perform this testing during a plant shutdown to prevent a reactor trip from occurring.

REFERENCES 1. Atomic Industrial Forum (AIF) GDC 15, Issued for Comment July 10, 1967.

2. UFSAR, Chapter 7.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.
5. IEEE-279-1971.
6. EWR-5126, "Guidelines For Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

(168)

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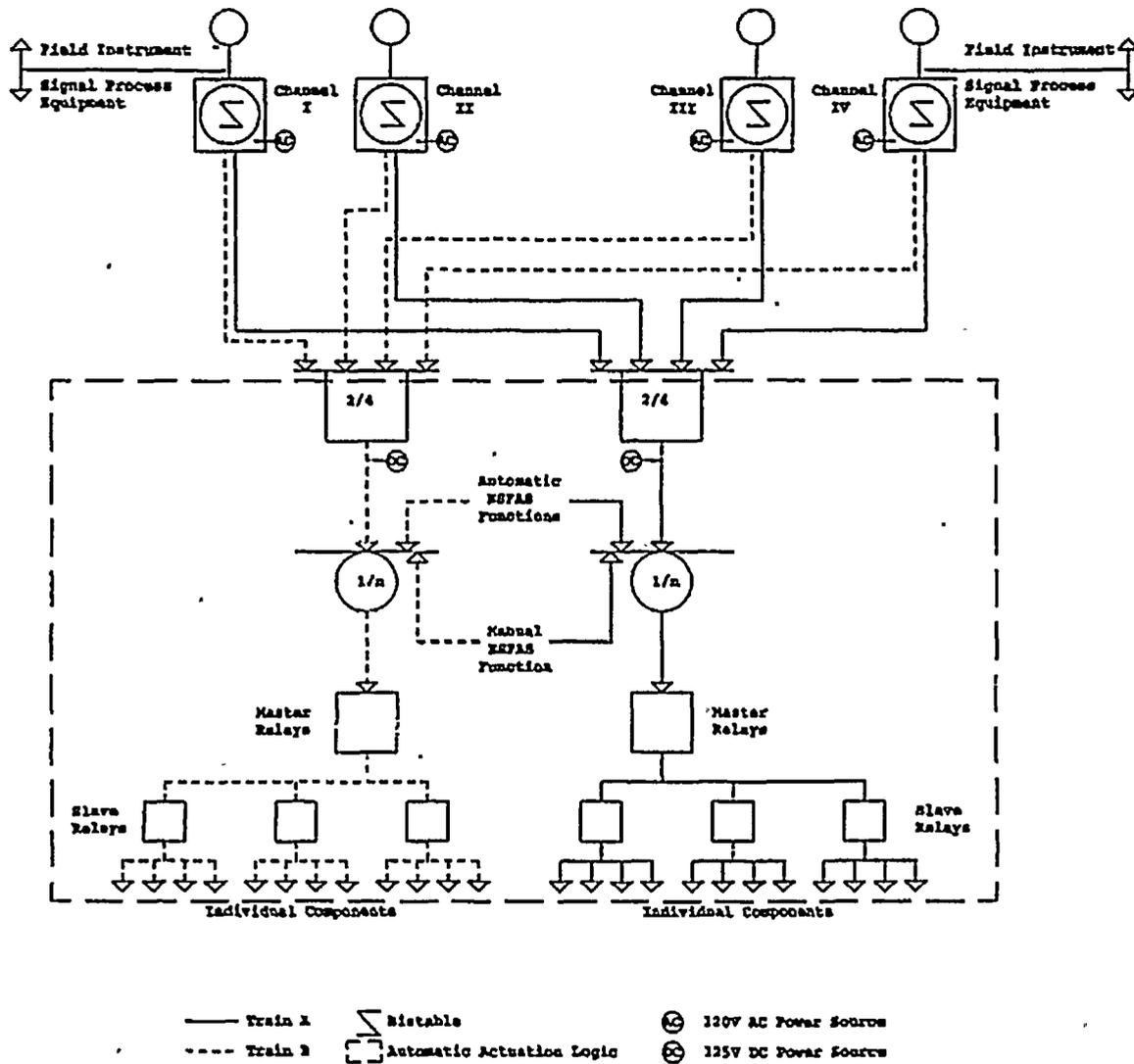


Figure B 3.3.2-1

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident conditions.

WCAP-14333, May 1995.

No changes

(continued)

D

BASES

(continued)

BASES

~~PAM Instrumentation~~
~~B 3.3.3~~

~~B 3.3 INSTRUMENTATION~~

~~B 3.3.3 Post Accident Monitoring (PAM) Instrumentation~~

BASES

No changes
~~BACKGROUND~~ ~~The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident conditions. This instrumentation provides the necessary support for the operator to take required manual actions, verify that automatic and required manual safety functions have been completed, and to determine if fission product barriers have been breached following a Design Basis Accident (DBA).~~

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior during an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified in Reference 1 addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO provide information for key parameters identified during implementation of Regulatory Guide 1.97 as Category I variables. Category I variables are organized into four types and are the key variables deemed risk significant because they are needed to:

- a. Provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is

(continued)



BASES

provided, and that are required for safety systems to accomplish their safety functions for DBAs (Type A).

- b. Provide the primary information required for the control room operator to verify that required automatic and manually controlled functions have been accomplished (Type B);

(continued)

BASES

BACKGROUND
(continued)

- c. Provide information to the control room operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release (Type C); and
- d. Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat (Type E).

All Type A and key Type B, C, and E parameters have been identified as Category I variables in Reference 1 which also provides justification for deviating from the NRC proposed list of Category I variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE
SAFETY ANALYSES

The PAM instrumentation ensures the availability of Regulatory Guide 1.97 Category I variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures for the primary success path of DBAs (e.g., loss of coolant accident (LOCA));
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether required automatic and manual safety functions have been accomplished;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category I, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A, variables are important for reducing public risk and ~~safety~~ satisfy Criterion 4.

(169)

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the plant Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident.

This LCO requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from obtaining the information necessary to determine the safety status of the plant, and to bring the plant to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. More than two channels may be required if failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

Table 3.3.3-1 lists all Category I variables identified by Reference 1.

(continued)

BASES

LCO
(continued)

Category I variables are considered OPERABLE when they are capable of providing immediately accessible display and continuous readout in the control room. The Hydrogen Monitors are considered OPERABLE when continuous readout is available in the Control Room or in the relay room. Each channel must also be supplied by separate electrical trains except as noted below. In addition, in accordance with LCO 3.0.6, it is not required to declare a supported system inoperable due to the inoperability of the support system (e.g., electric power). Since the inoperability of Instrument Bus D does not have any associated Required Actions, the loss of this power source may affect the OPERABILITY of the Pressurizer Pressure and SG Water Level (Narrow Range) Functions.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

1. Pressurizer Pressure

Pressurizer Pressure is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer pressure is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of pressure transmitters comprise the two channels required for this function:

- PT-429 and PT-431;
- PT-430 and PT-431;
- PT-429 and PT-449;
- PT-430 and PT-449; or
- PT-431 and PT-449

The loss of Instrument Bus D requires declaring PT-449 inoperable.

(continued)



BASES

LCO
(continued)

2. Pressurizer Level

Pressurizer Level is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer water level is also used to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of level transmitters comprise the two channels required for this function:

- LT-426 and LT-428; or
- LT-427 and LT-428.

3, 4: Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures are Category I variables (RCS Cold Leg Temperature is also a Type A variable) provided for verification of core cooling and long term surveillance of RCS integrity.

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify natural circulation in the RCS.

Temperature inputs are provided by two independent temperature sensor resistance elements and associated transmitters in each loop. Temperature elements TE-409B-1 and TE-410B-1 provide the required RCS cold leg temperature input for RCS Loops A and B, respectively. Temperature elements TE-409A-1 and TE-410A-1 provide the required RCS hot leg temperature input for RCS Loops A and B, respectively.

(continued)

BASES

LCO
(continued)

5. RCS Pressure (Wide Range)

RCS wide range pressure is a Type A variable provided for verification of core cooling and the long term surveillance of RCS integrity.

RCS pressure is used to verify delivery of SI flow to the RCS from at least one train when the RCS pressure is below the SI pump shutoff head. RCS pressure is also used to verify closure of manually closed pressurizer spray line valves and pressurizer power operated relief valves (PORVs) and for determining RCS subcooling margin.

RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and stop the residual heat removal pumps (RHR);
- to manually restart the RHR pumps;
- as reactor coolant pump (RCP) trip criteria;
- to make a determination on the nature of the accident in progress and where to go next in the emergency operating procedure; and
- to determine whether to operate the pressurizer heaters.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization.

(continued)

BASES

LCO

5. RCS Pressure (Wide Range) (continued)

RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication.

RCS pressure transmitters PT-420 and PT-420A provide the two required channels for this function.

6. RCS Subcooling Monitor

RCS Subcooling Monitor is a Type A variable provided for verification of core cooling and long term surveillance of RCS integrity. The RCS Subcooling Monitor is used to provide information to the operator, derived from RCS hot leg temperature and RCS pressure, on subcooling. RCS subcooling margin is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

The emergency operating procedures determine RCS subcooling margin based on the core exit thermocouples (CETs) and RCS pressure. Therefore, any of the following combination of parameters comprise the two required channels for this function:

- TI-409A and TI-410A; or
- One pressurizer pressure transmitter and two CETs in each of the four quadrants supplied by electrical train A and train B (i.e., total of two pressurizer pressure transmitters and 16 CETs).

(continued)



BASES

(continued)

BASES

LCO
(continued)

7. Reactor Vessel Water Level

Reactor Vessel Water Level is a Type A variable provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

When both RCPs are stopped, the Reactor Vessel Water Level Indication System (RVLIS) provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. When the RCPs are operating, RVLIS indicates the fluid fraction of the RCS. Measurement of the collapsed water level or fluid fraction is selected because it is a direct indication of the water inventory.

Level transmitters LT-490A and LT-490B provide the two required channels for this function.

(109)

8. Containment Sump B Water Level

Containment Sump B Water Level is a Type A variable provided for verification and long term surveillance of RCS integrity.

Containment Sump B Water Level is used to determine:

- containment sump level for accident diagnosis;
- when to begin the recirculation procedure; and
- whether to terminate SI, if still in progress.

Level transmitters LT-942 and LT-943, each with five discrete level switches, provide the two required channels for this function.

(continued)

BASES

LCO
(continued)

9. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is a Type A variable provided for verification of RCS and containment OPERABILITY.

Containment Pressure (Wide Range) is used to determine the type of accident in progress and when, and if, to use emergency operating procedure containment adverse values.

Any of the following combinations of pressure transmitters comprise the two required channels for this function:

- PT-946 and PT-948; or
- PT-950 and PT-948.

10. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) is a Type E, Category I variable provided to monitor for the potential of significant radiation releases into containment and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Containment radiation level is used to determine the type of accident in progress (e.g., LOCA), and when, or if, to use emergency operating procedure containment adverse values.

Radiation monitors R-29 and R-30 are used to provide the two required channels for this function.

11. Hydrogen Monitors

Hydrogen Concentration is a Type C Category I variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

(continued)



BASES

LCO

11. Hydrogen Monitors (continued)

Hydrogen monitors HMSLCPA and HMSLCPB provide the two required channels for this function. In addition, the Post Accident Sampling System may take the place of one of these monitors. The PASS system Hydrogen Function is not required to provide continuous readout in the control room or relay room for OPERABILITY.

12. Condensate Storage Tank (CST) Level

CST Level is a Type A variable provided to ensure a water supply is available for the preferred Auxiliary Feedwater (AFW) System. The CST consists of two identical tanks connected by a common outlet header.

CST level is used to determine:

- if sufficient CST inventory is available immediately following a loss of normal feedwater or small break LOCA; and
- when to manually replenish the CST or align the safety related source of water (service water) to the preferred AFW system.

Level transmitters LT-2022A and LT-2022B provide the two required channels for this function. However, only the level transmitter associated with the CST(s) required by LCO 3-7-6, "Condensate Storage Tank(s)" are required for this LCO.

(6)

13. Refueling Water Storage Tank (RWST) Level

RWST Level is a Type A variable provided for verifying a water source to the SI, RHR, and CS Containment Spray (CS) Systems.

(16)

(continued)

BASES

~~The RWST level accuracy is established to allow an adequate supply of water to the SI, RHR, and CS pumps during the switchover to the recirculation phase of an accident.~~ LCO 13. ~~A high degree of~~ Refueling Water Storage Tank (RWST) Level (continued)

~~The RWST level accuracy is required established to maximize the time available to the operator to complete the switchover to the sump recirculation phase and ensure sufficient water is available to maintain an adequate NPSH to operating supply of water to the SI, RHR, and CS pumps during the switchover to the recirculation phase of an accident. A high degree of accuracy is required to maximize the time available to the operator to complete the switchover to the sump recirculation phase and ensure sufficient water is available to maintain adequate net positive suction head (NPSH) to operating pumps.~~

(169)

Level transmitters LT-920 and LT-921 provide the two required channels for this function.

LCO—14. RHR Flow

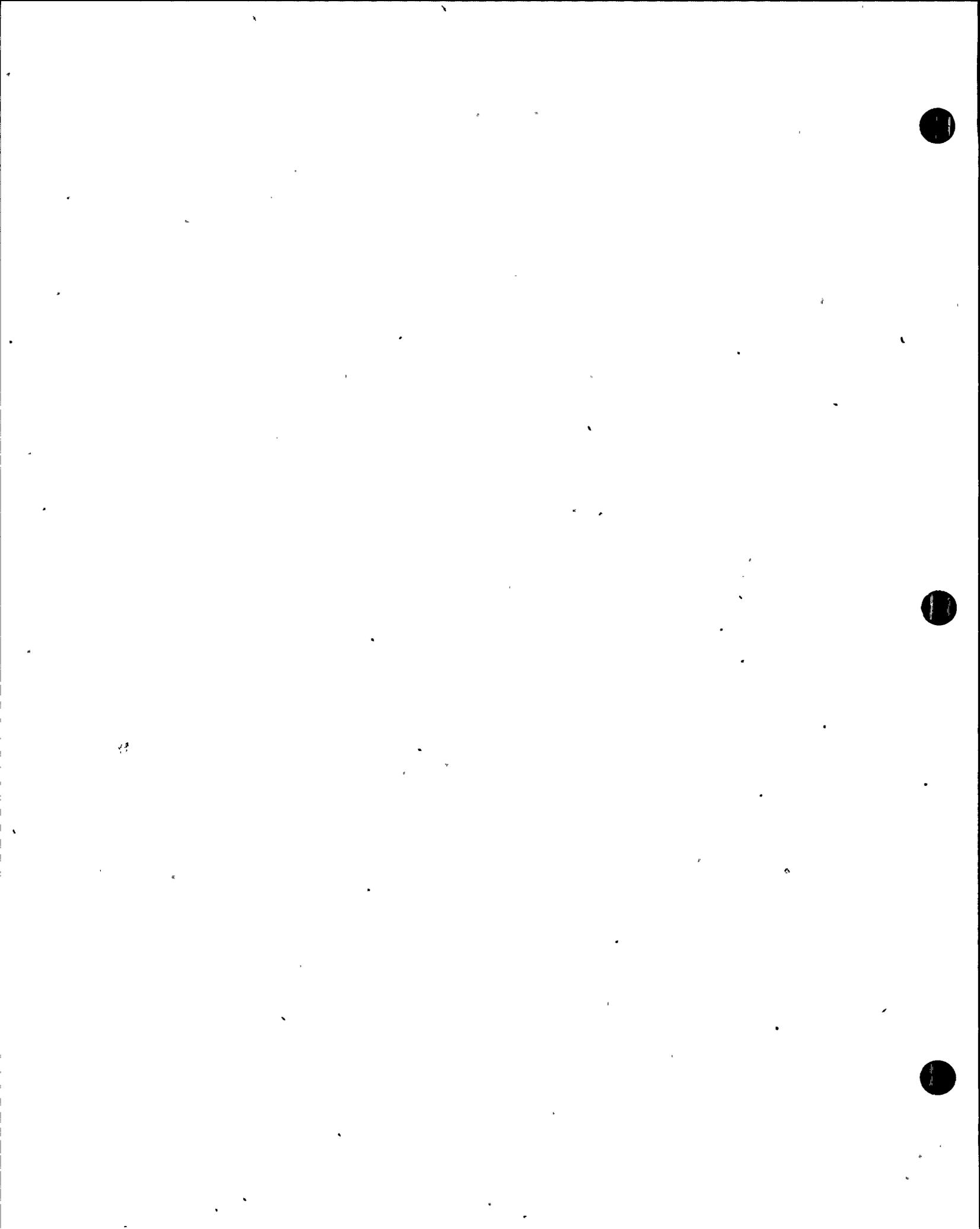
—(continued)

RHR Flow is a Type A variable provided for verifying low pressure safety injection to the reactor vessel and to the CS and SI pumps.

RHR flow is used to determine when to stop the RHR pumps and if sufficient flow is available to the CS and SI pumps during recirculation.

Since different flow transmitters are used to verify injection to the reactor vessel and to verify flow to the CS and SI pumps, FT-626 and FT-931A comprise one required channel and FT-689 and FT-931B comprise a second required channel.

(continued)



BASES

~~15, 16, 17, 18. Core Exit Temperature~~

~~Core Exit Temperature is a Type A variable provided for verification and long term surveillance of core cooling. LCO~~

~~15, 16, 17, 18. Core Exit Temperature
(continued)~~

No change

Core Exit Temperature is a type A variable provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid CETs necessary for measuring core cooling. The evaluation determined the necessary complement of CETs required to detect initial core recovery and trend the ensuing core heatup. The evaluation accounted for core nonuniformities, including incore effects of the radial decay power distribution, excore effects of reflux in the hot legs, and nonuniform inlet temperatures. Based on these evaluations, adequate core cooling is ensured with two valid Core Exit Temperature channels per quadrant with two CETs per required channel.

Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for plant stabilization and cooldown control.

(continued)



BASES

LC0

~~15, 16, 17, 18. Core Exit Temperature (continued)~~

~~Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core.~~

No change

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Because of the small core size, two randomly selected thermocouples are sufficient to meet the two thermocouples per channel requirement in any quadrant. However, a CET which lies directly on the dividing line between two quadrants can only be used to satisfy the minimum required channels for one quadrant.

A CET is considered OPERABLE when it is within $\pm 35^\circ\text{F}$ of the average CET reading ~~except for the CETs associated with peripheral assemblies. At least two~~ These CETs from each of the following trains must be OPERABLE in each of the four quadrants:

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Train A		Train B	
CET	Location	CET	Location
T2	M6	T11	I4
T5	J3	T3	L7
T6	I2	T4	K3
T7	J6	T10	J9
T8	L10	T13	K11
T9	J8	T14	D12
T12	H6	T16	H10
T15	H9	T17	E10
T18	F8	T19	G7
T21	C11	T20	C8
T22	(A7, B5, C3, C)		

AFW Flow

(continued)

169

AFW Flow is a Type A variable provided to monitor operation. ~~11, D2, D12, H13, I2, K3, K11, L10, and M6~~ are considered OPERABLE when they are within $\pm 43^\circ\text{F}$ of the preferred AFW system average CET reading.

(continued)

BASES

169

~~The AFW System provides decay heat removal via At least two CEIs from each of the SGs and is comprised following trains must be OPERABLE in each of the preferred AFW System and the Standby AFW (SAFW) System four quadrants.~~

(continued)

BASES

ECO

15, 16, 17, 18.

Core Exit Temperature (continued)

Train A		Train B	
GET	Location	GET	Location
T2	M6	T1	L4
T5	J3	T3	L7
T6	I2	T4	K3
T7	J6	T10	J9
T8	L10	T13	K11
T9	J8	T14	D12
T12	H6	T16	H10
T15	H9	T17	E10
T18	F8	T19	G7
T21	C11	T20	G8
T22	H11	T24	F12
T23	H13	T25	G12
T26*	I10	T27	E6
T28	D5	T29	E4
T33	D2	T30*	G4
T34	C3	T31	G2
T36	B7	T32*	G1
T38	B5	T35	A7
T39	D7	T37	G6

* - These thermocouples are in the reactor vessel head and cannot be credited with respect to this ECO.

169

169

19 20
18, 19

AFW Flow

AFW Flow is a Type A variable provided to monitor operation of the preferred AFW system.

No change

(continued)

BASES

(169)
LCO 18, 19, 20

AFW Flow (continued)

No change

The AFW System provides decay heat removal via the SGs and is comprised of the preferred AFW System and the Standby AFW (SAFW) System. The use of the preferred AFW or SAFW System to provide this decay heat removal function is dependent upon the type of accident. AFW flow indication is required from the three pump trains which comprise the preferred AFW System since these pumps automatically start on various actuation signals. The failure of the preferred AFW System (e.g., due to a high energy line break (HELB) in the Intermediate Building) is detected by AFW flow indication. At this point, the SAFW System is manually aligned to provide the decay heat removal function.

SAFW flow can also be used to verify that AFW flow is being delivered to the SGs. However, the primary indication of this is provided by SG water level. Therefore, flow indication from the SAFW pumps is not required.

Each of the three preferred AFW pump trains has two redundant transmitters; however, only the flow transmitter supplied power from the same electrical train as the AFW pump is required for this LCO. Therefore, flow transmitters FT-2001 (MCB indicator FI-2021A) and FT-2007 (MCB indicator FI-2024A) comprise the two required channels for SG A and FT-2002 (MCB indicator FI-2022A) and FT-2006 (MCB indicator FI-2023A) comprise the two required channels for SG B.

(169)

(continued)

BASES

LCO

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(continued)

20, 21, 22, 23, 24.

SG Water Level (Narrow and Wide Range)

SG Water Level is a Type A variable provided to monitor operation of decay heat removal via the SGs. For the narrow range level, the signals from the transmitters are independently indicated on the main control board as 0% to 100%. This corresponds to approximately above the top of the tube bundles to the top of the swirl vane separators (span of 143 inches). For the wide range level, signals from the transmitters are indicated as 0 to 520 inches (0% to 100%) on the main control board.

SG Water Level (Narrow and Wide Range) is used to:

- identify the faulted SG following a tube rupture;
- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify an SGTR); and
- verify plant conditions for termination of SI during secondary plant HELBs outside containment.

Redundant monitoring capability is provided by two trains of instrumentation per SG.

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~~S/GSG Water Level (Narrow Range) requires 2 channels of indication (one per SG) which can be met using any of the following combinations of level transmitters:~~

- ~~• LT 461 and LT 471;~~
- ~~• LT 462 and LT 471;~~
- ~~• LT 462 and LT 472;~~
- ~~• LT 462 and LT 473;~~
- ~~• LT 463 and LT 472; and~~

(continued)

BASES

(235)

• ~~LT 463 and LT 473~~SG.

(continued)

BASES

This can be met using any of the following combinations of level transmitters for SG A:

235

- LT-461 and LT-462;
- LT-462 and LT-463; or
- LT-461 and LT-463;

(continued)

BASES

LCO

~~20,~~ 21, 22, 23, 24.
SG Water Level (Narrow and Wide Range) (continued)

~~The loss of Instrument Bus D requires declaring LT-463 and LT-471 inoperable for SG B; any of the following combinations of level transmitters can be used:~~

- ~~LT-471 and LT-473;~~
- ~~LT-471 and LT-472; or~~
- ~~LT-472 and LT-473.~~

(235)

~~The loss of Instrument Bus D requires declaring LT-463 and LT-471 inoperable.~~

SG Water Level (Wide Range) requires 2 channels of indication per SG. Two channels per SG are required since the loss of one channel with no backup available may result in the complete loss of information required by the operators to accomplish necessary safety functions. Level transmitters LT-504 and LT-505 comprise the two required channels for SG A and LT-506 and LT-507 comprise the two required channels for SG B.

~~2225,~~ 26. SG Pressure

SG Pressure is a Type A variable provided to monitor operation of decay heat removal via the SGs. The signals from the transmitters are calibrated for a range of 0 psig to 1400 psig. Redundant monitoring capability is provided by three available trains of instrumentation.

Any of the following combinations of pressure transmitters comprise the two required channels for ~~this function~~ SG A:

(235)

- ~~PT-468 and PT-478~~ ~~PT-482;~~ or
- PT-469 and PT-478;
- ~~PT-479 and PT-482;~~ and

(continued)

BASES

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• — ~~PT 482 and PT 483~~ PT 482.

(continued)



BASES

APPLICABILITY ~~The PAM instrumentation LCO is applicable in MODES 1, 2, 25 and 3 26.~~ ~~SG Pressure (continued)~~

Any of the following combinations of pressure transmitters comprise the two required channels for SG B:

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• PT-479 and PT-478; or

• PT-478 and PT-483.

APPLICABILITY The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, the PAM instrumentation is not required to be OPERABLE because plant conditions are such that the likelihood of an event that would require PAM instrumentation is low.

ACTIONS

The ACTIONS are modified by two Notes.

Note 1 has been added to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)



BASES

(continued)

ACTIONS A.1

~~Condition A applies to all PAM instrumentation functions when one or more Functions have one required channel that is inoperable. Condition Required Action A addresses the situation where one or more required channels for one or more required Functions are inoperable. The Required Action is to immediately refer to Table 3.3.3-1 and take the Required Actions for the instrumentation functions affected. This requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (or critical automatic action is assumed to occur from the referenced Conditions and Required Actions these instruments), and the low probability of an event requiring PAM Instrumentation during this interval.~~

~~ACTIONS — B Condition A is modified by a Note which states that the Condition is not applicable to Table 3.3.3-1 Functions 3 and 4.~~

~~These Functions are addressed by Condition B applies when one C which provides the necessary required actions for these single channel is inoperable Functions.~~

~~This Condition includes the inoperability of one RCS hot leg or cold leg temperature channel B. It also includes:~~

~~Condition B applies when the inoperability of one SG Water Level (Wide Range) channel in one or both SGs Required Action and associated Completion Time for Condition A is not met.~~

~~Required Action B This Condition requires the immediate initiation of actions to prepare and submit a special report to the NRC. It requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other~~

(continued)

BASES

~~non Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.~~

6.1

~~Condition C applies when the Required Action and associated Completion Times for Condition B is not met. This Condition requires the immediate initiation of actions to prepare and submit a Special Report to the NRC. This report shall be submitted within the following 14 days from the time the Condition is entered. This report shall discuss the results of the root cause evaluation of the inoperability and identify proposed restorative actions or alternate means of providing the required function. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation. If alternate means are to be used, they must be developed and tested prior to submittal of the Special Reports~~ special report.

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(continued)

BASES

ACTIONS D-1
—(continued)

Condition D applies when a Function has two inoperable required channels or when a Function has one inoperable required channel and no diverse channel OPERABLE (i.e., complete loss of RCS Hot Leg Temperature or RCS Cold Leg Temperature Functions). This Condition includes the inoperability of two SG Water Level (Wide Range) to one or both SGs. This Condition requires restoring one channel in the affected Function to OPERABLE status within 7 days.

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(continued)

BASES

ACTIONS **C.1**
(continued)

Condition C applies when a Function has one inoperable required channel and no diverse channel OPERABLE (i.e., loss of RCS Hot Leg Temperature or RCS Cold Leg Temperature Functions). This Condition requires restoring the inoperable channel in the affected Function to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with either two required channels inoperable in a Function or complete loss of function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of the inoperable channel limits the risk that the PAM Function will be in a degraded condition should an accident occur.

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Condition C is modified by a Note which states that this Condition is only applicable to Table 3.3.3-1 Functions 3 and 4. All remaining Functions are addressed by Condition A with one channel inoperable.

D.1

Condition D applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action D.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

(continued)



BASES

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^D
Condition C is modified by a Note that excludes Function 11 since the inoperability of two hydrogen monitor channels is addressed by Condition E.

(continued)

BASES

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~~Condition E ACTIONS~~
~~E applies when two hydrogen monitor channels are~~
~~inoperable.]~~

(continued)

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Condition E applies when two hydrogen monitor channels are inoperable. This Condition requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System ~~or the redundant hydrogen monitor channel~~ to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA which would potentially require use of the hydrogen recombiners would occur during this time.

(continued)



BASES

ACTIONS F.1

(continued)

If one channel for Function 7 or 10 cannot be restored to OPERABLE status within the required Completion Time of Condition D, the plant must take immediate action to prepare Condition F applies when the Required Action and submit a Special Report to the NRC associated Completion Time of Condition C, D, or E are not met. Required Action F requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C, D, or E, and the associated Completion Time has expired, Condition F is entered for that channel and provides for transfer to the appropriate subsequent Condition. This report shall be submitted within the following 14 days from the time the action is required. This report discusses the alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation, the degree to which the alternate means are equivalent to the installed PAM channels, the areas in which they are not equivalent, and a schedule for restoring the normal PAM channels.

These alternate means must have been developed and tested and may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time.

G.1 and G.2

If one channel for Function 3 and 4 cannot be restored to OPERABLE status within the required Completion Time for Condition C, if one channel for Function 1, 2, 3, 4, 5, 6, 8, 9, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, or 22, 23, 24, 25, or 26 cannot be restored to OPERABLE status within the required Completion Time of Condition D, or if one channel for Function 11 cannot be restored to OPERABLE status within the required Completion Time of Condition E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are

(continued)



BASES

reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

~~Condition G is modified by a Note which clarifies that this Condition is not applicable to Functions 7 and 10~~ ACTIONS

H.

(235)

(continued)

BASES

~~SURVEILLANCE~~ ~~A Note has been added to the SR Table to clarify that~~
~~REQUIREMENTS~~ ~~SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM~~
~~instrumentation.~~

(continued)

If one channel for Function in Table 3.3.3-17 or 10 cannot be restored to OPERABLE status within the required Completion Time of Condition D, the plant must take immediate action to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the action is required. This report discusses the alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation, the degree to which the alternate means are equivalent to the installed PAM channels, the areas in which they are not equivalent, and a schedule for restoring the normal PAM channels.

These alternate means must have been developed and tested and may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time.

~~SURVEILLANCE~~ ~~A Note has been added to the SR Table to clarify that~~
~~REQUIREMENTS~~ ~~SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation~~
~~Function in Table 3.3.3-1~~

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation

(continued)

BASES

instrumentation should be compared to similar plant instruments located throughout the plant.

(continued)

BASES

No change.

SURVEILLANCE

SR 3.3.3.1 (continued)

Channel check acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

(continued)

BASES

SURVEILLANCE ——— SR 3.3.3.2

REQUIREMENTS
(continued) ———

(169) A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. UFSAR, Section 7.5.2.
 2. Regulatory Guide 1.97, Rev. 3.
 3. NUREG-0737, Supplement 1, "TMI Action Items."
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B 3.3 INSTRUMENTATION

B 3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

(20) The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe plant operation. The LOP DG start instrumentation consists of two channels on each of safeguards Buses 14, 16, 17, and 18 (Ref. 1). Each channel contains one loss of voltage relay and one degraded voltage relay (see Figure B 3.3.4-1). A one-out-of-two logic in both channels will cause the following actions on the associated safeguards bus:

- a. trip of the normal feed breaker from offsite power;
- b. trip of the bus-tie breaker to the opposite electrical train (if closed);
- c. shed of all bus loads except the CS pump, component cooling water pump (if no safety injection signal is present), and safety related motor control centers; and
- d. start of the associated DG.

The degraded voltage logic is provided on each 480 V safeguards bus to protect Engineered Safety Features (ESF) components from exposure to long periods of reduced voltage conditions which can result in degraded performance and to ensure that required motors can start. The loss of voltage logic is provided on each 480 V safeguards bus to ensure the DG is started within the time limits assumed in the accident analysis to provide the required electrical power if offsite power is lost.

The degraded voltage relays have time delays which have inverse operating characteristics such that the lower the bus voltage, the faster the operating time. The loss of voltage relays have definite time delays which are not related to the rate of the loss of bus voltage. These time delays are set to permit voltage transients during worst case motor starting conditions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The LOP DG start instrumentation is required for the ESF Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS). Undervoltage conditions which occur independent of any accident conditions result in the start and bus connection of the associated DG, but no automatic loading occurs.

Accident analyses credit the loading of the DG based on the loss of offsite power during a Design Basis Accident (DBA). The most limiting DBA of concern is the large break loss of coolant accident (LOCA) which requires ESF Systems in order to maintain containment integrity and protect fuel contained within the reactor vessel (Ref. 2). The detection and processing of an undervoltage condition, and subsequent DG loading, has been included in the delay time assumed for each ESF component requiring DG supplied power following a DBA and loss of offsite power.

The loss of offsite power has been assumed to occur either coincident with the DBA or at a later period (40 to 90 seconds following the reactor trip) due to a grid disturbance caused by the turbine generator trip. If the loss of offsite power occurs at the same time as the safety injection (SI) signal parameters are reached, the accident analyses assumes the SI signal will actuate the DG within 2 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12 seconds total time). If the loss of offsite power occurs before the SI signal parameters are reached, the accident analyses assumes the LOP DG start instrumentation will actuate the DG within 2.75 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12.75 seconds total time). If the loss of offsite power occurs after the SI signal parameters are reached (grid disturbance), the accident analyses assumes the LOP DG start instrumentation will open the feeder breaker to the affected bus within 2.75 seconds and the DG will connect to the bus within an additional 1.5 seconds (DG was actuated by SI signal). The grid disturbance has been evaluated based on a 140°F peak clad temperature penalty during a LOCA and demonstrated to result in acceptable consequences.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The degraded voltage and undervoltage setpoints are based on the minimum voltage required for continued operation of ESF Systems assuming worst case loading conditions (i.e., maximum loading upon DG sequencing). The Trip Setpoint for the loss of voltage relays, and associated time delays, have been chosen based on the following considerations:

- a. Actuate the associated DG within 2.75 seconds as assumed in the accident analysis; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest assumed voltage drop below the voltage setting and the reset value of the trip function.

The Trip Setpoint for the degraded voltage channels, and associated time delays, have been chosen based on the following considerations;

- a. Prevent motors supplied by the 480 V bus from operating at reduced voltage conditions for long periods of time; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available, and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest voltage drop below the maximum voltage setting and the reset value of the trip function.

The LOP DG start instrumentation channels satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

This LCO requires that each 480 V safeguards bus have two OPERABLE channels of the LOP DG start instrumentation in MODES 1, 2, 3, and 4 when the associated DG supports safety systems associated with the ESFAS. In MODES 5 and 6, the LOP DG start instrumentation channels for each 480 V safeguards bus must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents.

The LOP start instrumentation is considered OPERABLE when two channels, each comprised of one degraded voltage and one loss of voltage relays are available for each 480 V safeguards bus (i.e., Bus 14, 16, 17, and 18). Each of the LOP channels must be capable of detecting undervoltage conditions within the voltage limits and time delays assumed in the accident analysis.

The Allowable Values and Trip Setpoints for the degraded voltage and loss of voltage Functions are specified in SR 3.3.4.2. The Allowable Values specified in SR 3.3.4.2 are those setpoints which ensure that the associated DG will actuate within 2.75 seconds on undervoltage conditions, and that the DG will not actuate on momentary voltage drops which could affect ESF actuation times as assumed in the accident analysis. The Trip Setpoints specified in SR 3.3.4.2 are the nominal setpoints selected to ensure that the setpoint measured by the Surveillance does not exceed the Allowable Value accounting for maximum instrument uncertainties between scheduled surveillances. Therefore, LOP start instrumentation channels are OPERABLE when the CHANNEL CALIBRATION "as left" value is within the Trip Setpoint limits and the CHANNEL CALIBRATION and TADOT "as found" value is within the Allowed Value setpoints. The basis for all setpoints is contained in Reference 3.

(continued)

BASES

APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the 480 V safeguards buses.

ACTIONS In the event a relay's Trip Setpoint is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then the channel must be declared inoperable and the LCO Condition entered as applicable.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. This Note states that separate Condition entry is allowed for each 480 V safeguards bus.

A.1

~~Condition A applies to the LOP DG start Function with one or more 480 V bus(es) with one channel per bus inoperable. Required Action A.~~

(216)

~~With one channel requires the inoperable. Required Action A channel(s) to be placed in trip within 6 hours. 1 requires that channel to be placed in trip within 6 hours. With an undervoltage channel in the tripped condition, the LOP DG start instrumentation channels are configured to provide a one-out-of-one logic to initiate a trip of the incoming offsite power for the respective bus. The remaining OPERABLE channel is comprised of one-out-of-two logic from the degraded and loss of voltage relays. Any additional failure of either of these two OPERABLE relays requires entry into Condition B.~~

(continued)



BASES

ACTIONS
(continued)

B.1

(216) Condition B applies to the LOP DG start Function when the Required Action and associated Completion Time for Condition A are not met or when ~~with one or more 480 V bus(es) with two channels of LOP start instrumentation per bus are inoperable.~~

Condition B requires immediate entry into the Applicable Conditions specified in LCO 3.8.1, "AC Sources—MODES 1, 2, 3, and 4," or LCO 3.8.2, "AC Sources—MODES 5 and 6," for the DG made inoperable by failure of the LOP DG start instrumentation. The actions of those LCOs provide for adequate compensatory actions to assure plant safety.

SURVEILLANCE
REQUIREMENTS

(109) The Surveillances are modified by a Note to indicate that, when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 64 hours, provided the second channel maintains trip capability. Upon completion of the Surveillance, or expiration of the 4 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on ~~the~~ assumption that 4 hours is the average time required to perform channel surveillance. Based on engineering judgement, ~~the~~ 4 hour testing allowance does not significantly reduce the probability that the LOP DG start instrumentation will trip when necessary.

SR 3.3.4.1

This SR is the performance of a TADOT every 31 days. This test checks trip devices that provide actuation signals directly. For these tests, the relay Trip Setpoints are verified and adjusted as necessary to ensure Allowable Values can still be met. The 31 day Frequency is based on the known reliability of the relays and controls and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

214

SR 3.3.4.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling of the LOP DG start instrumentation for each 480 V bus.

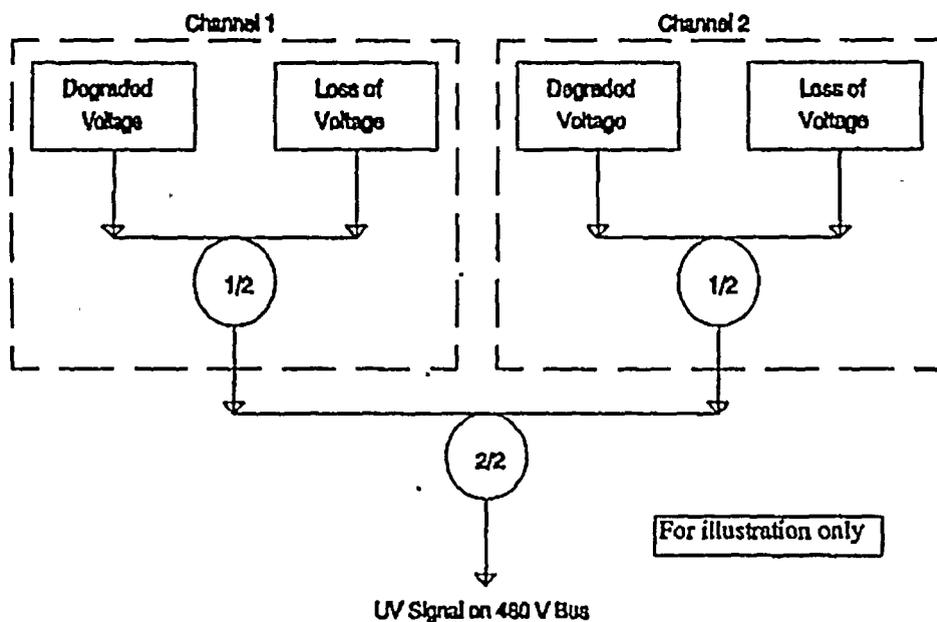
The voltage setpoint verification, as well as the time response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience consistent with the typical industry refueling cycle and is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 8.3.
 2. UFSAR, Chapter 15.
 3. RG&E Design Analysis DA-EE-93-006-08, "480 Volt Undervoltage Relay Settings and Test Acceptance Criteria."
-



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Figure B 3.3.4-1
DG LOP Instrumentation

Figure B 3.3.4-1
DG LOP Instrumentation

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~~Containment Ventilation Isolation~~ Instrumentation
B 3.3.5

B 3.3 INSTRUMENTATION

~~B 3.3.5 Control Room Emergency Air Treatment System (CREATS) Actuation~~
~~Containment Ventilation Isolation Instrumentation~~

BASES

BACKGROUND

~~The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. Containment ventilation isolation instrumentation closes the containment isolation valves in the Mini-Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-Purge System may be used in all MODES while the Shutdown Purge System may only be used with the reactor shutdown.~~

~~Containment ventilation isolation initiates on a containment isolation signal, containment radiation signal, or by manual actuation of containment spray (CS). The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss the containment isolation and manual containment spray modes of initiation.~~

~~Two ^{containment} radiation monitoring channels are ~~also~~ provided as input to the containment ventilation isolation. The two radiation detectors are of different types: gaseous (R-12), and particulate (R-11). Both detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the two channels are not considered redundant. Instead, they are treated as two one-out-of-one Functions. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.~~

(continued)

221
BASES

BACKGROUND
(continued) The Mini-Purge System has inner and outer containment isolation valves in its supply and exhaust ducts while the Shutdown Purge System only has one valve located outside containment since the inside valve was replaced by a blind flange that is used during MODES 1, 2, 3, and 4. A high radiation signal from any one of the two channels initiates containment ventilation isolation, which closes all isolation valves in the Mini-Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABLE SAFETY ANALYSES The safety analyses assume that the containment remains intact with penetrations unnecessary for accident mitigation functions isolated early in the event, within approximately 60 seconds. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the containment isolation signal to ensure closing of the ventilation valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown even though containment isolation is not specifically credited for this event. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accident⁰ offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.5-1, is OPERABLE.

(continued)

221

BASES

LCO 1.
(continued)

Manual Initiation

The LCO requires two channels to be OPERABLE. The operator can initiate Containment Ventilation Isolation at any time by using either of two switches in the control room. Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.

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Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 2, Containment Spray - Manual Initiation, and ESFAS Function 3, Containment Isolation. The applicable MODES and specified conditions for the containment ventilation isolation portion of these Functions are different and less restrictive than those for their respective CS and ESFAS roles. If one or more of the CS or containment isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their respective isolation Functions in LCO 3.3.2 need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

(continued)

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BASES

LCO ²
_{3.2}
(continued)

Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur.

³
_{3.2}

Containment Isolation

Refer to LCO 3.3.2, Function 3, for all initiating Functions and requirements.

⁴
_{3.2}

Containment Spray—Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

APPLICABILITY

The Manual Initiations, Automatic Actuation Logic and Actuation Relays, Containment Isolation, Containment Spray—Manual Initiation, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

(continued)



(221)
BASES (continued)

ACTIONS The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel. Since the two containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 1-hour allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

(continued)

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BASES

ACTIONS **B.1 (continued)**

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place each valve in its closed position or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

(continued)

221

BASES

SURVEILLANCE REQUIREMENTS A Note has been added to the SR Table to clarify that Table 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every ²⁴ 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LGO required channels.

SR 3.3.5.2

A GOT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. The setpoint shall be left consistent with the current plant specific calibration procedure tolerance.

(continued)



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BASES

SURVEILLANCE REQUIREMENTS SR 3.3.5.3

(continued)

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 24 months. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.5.4

This SR is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 24 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.)

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

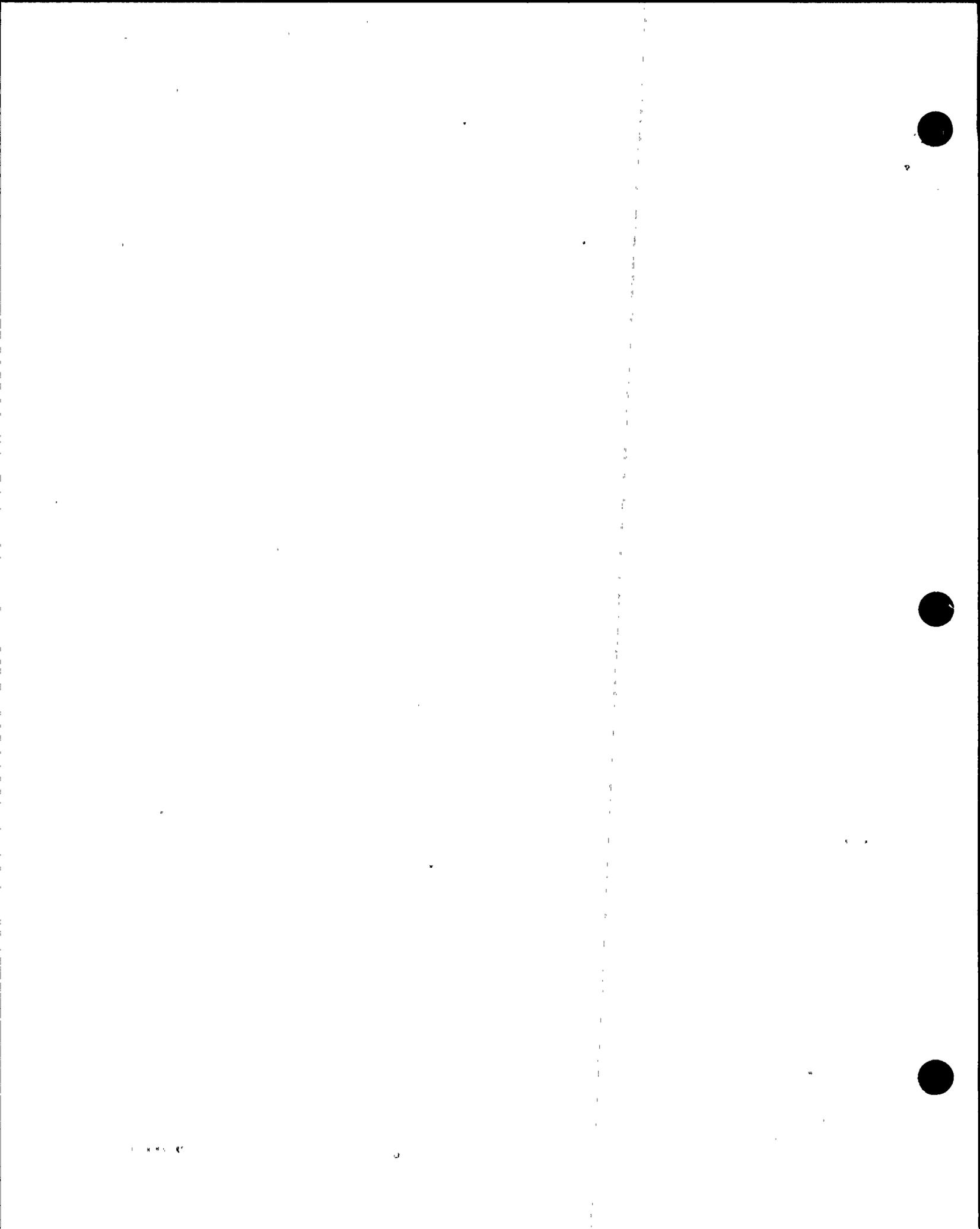
The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.5.5⁴

A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

(continued)



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BASES

- REFERENCES
1. 10 CFR 100.11
 2. NUREG-1366
-
-

B 3.3 INSTRUMENTATION

B 3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

221

BASES

BACKGROUND

The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This system is described in the Bases for LCO 3.7.9, "Control Room Emergency Air Treatment System (CREATS)." This LCO only addresses the actuation instrumentation for the high radiation state CREATS Mode F.

The high radiation state CREATS Mode F actuation instrumentation consists of noble gas (R-36), particulate (R-37), and iodine (R-38) radiation monitors. These detectors are located on the operating level on the Turbine Building and utilize a common air supply pump. A high radiation signal from any of these detectors will initiate the CREATS filtration train and isolate each air supply path with two dampers. The control room operator can also initiate the CREATS filtration train and isolate the air supply paths by using a manual pushbutton in the control room.

APPLICABLE SAFETY ANALYSES

The location of components and CREATS related ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4; as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 1). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator, and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

(continued)



BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

In MODES 5 and 6, and during movement of irradiated fuel-assemblies, the CREATS ensures control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident.

No change

~~SAFETY ANALYSES~~ ~~APPLICABLE~~ — The CREATS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

~~(continued)~~

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREATS is OPERABLE.

1. Manual Initiation

The LCO requires one train to be OPERABLE. The train consists of one pushbutton and the interconnecting wiring to the actuation logic. The operator can initiate the CREATS Filtration train at any time by using a pushbutton in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals required by this LCO.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires one train of Actuation Logic and Actuation Relays to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation system, including the initiation relay contacts responsible for actuating the CREATS.

3. Control Room Radiation Intake Monitor

The LCO specifies single channels of iodine (R-38), noble gas (R-36), and particulate (R-37) of the Control Room Intake Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREATS filtration train and isolation dampers remains OPERABLE.

(continued)

BASES

APPLICABILITY In MODES 1, 2, 3, and 4, the CREATS actuation instrumentation must be OPERABLE to control operator exposure during and following a Design Basis Accident.

No change

~~APPLICABILITY~~—In MODE 5 or 6, the CREATS actuation instrumentation is
~~(continued)~~— required to cope with the release from the rupture of a waste gas decay tank.

During movement of irradiated fuel assemblies, the CREATS actuation instrumentation must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

The most common cause of channel inoperability is failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. The "as left" Trip Setpoint must be within the tolerance specified by the calibration procedure. If the "as found" Trip Setpoint exceeds the limits specified in Table 3.3.5-1, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.5-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel/train of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies to one or more Functions with one ~~or~~
~~more~~ channels of the CREATS actuation instrumentation
inoperable.

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(continued)

BASES

ACTIONS

~~If one or more radiation monitor channels, the manual initiation train, or the automatic actuation logic train is inoperable, action must be taken to restore OPERABLE status within 1 hour or isolate the control room from outside air.1~~
(continued)

No change

~~If one or more radiation monitor channels, the manual initiation train, or the automatic actuation logic train is inoperable, action must be taken to restore OPERABLE status within 1 hour or isolate the control room from outside air. In this Condition for the manual initiation train inoperable or a radiation monitor channel inoperable, the remaining CREATS actuation instrumentation is adequate to perform the control room protection function but the actuation time or responsiveness of the CREATS may be affected. In this Condition for the automatic actuation logic train inoperable or all radiation monitor channels inoperable, the CREATS is not capable of performing its intended automatic function. This is considered a loss of safety function. The CREATS, however, may still be capable of being placed in CREATS Mode F by manual operator actions. The 1 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to automatically isolate the control room or be manually isolated by the operator.~~

The Required Action for Condition A is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

(continued)

BASES

ACTIONS B.1 and B.2

(continued)

No change

Condition B applies when the Required Action and associated Completion Time of Condition A has not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE that minimizes accident risk. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

ACTIONS ~~_____ C.~~

No Change

C.1, C.2, and C.3

—(continued)

(169)

Condition C applies when the Required Action and associated Completion Time of Condition A has not been met in MODE 5, or 6, or during movement of irradiated fuel assemblies. Actions must be initiated immediately to restore the inoperable channel(s) ~~or train~~ to OPERABLE status to ensure adequate isolation capability in the event of a waste gas decay tank rupture. Movement of irradiated fuel assemblies and CORE ALTERATIONS must also be suspended immediately to reduce the risk of accidents that would require CREATS actuation. This places the plant in a condition that minimizes risk. This does not preclude movement of fuel or other components to a safe position.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table ~~3.3.5-13~~ ~~3.6-1~~ determines which SRs apply to which CREATS Actuation Functions.

(221)

(continued)

BASES

SR ~~3.3.5.1~~

221

REQUIREMENTS
(continued)

~~SURVEILLANCE~~ SR ~~3.3.6.1~~

This SR is the performance of a COT once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

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SR ~~3.3.5.2~~ ~~3.3.6.2~~

This SR is the performance of a TADOT of the Manual Actuation Functions every 24 months. The Manual Actuation Function is tested up to, and including, the master relay coils.

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.3.5.2~~ (continued)
~~REQUIREMENTS~~

The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints associated with them.

(271)

~~SR 3.3.5.3~~ 3.6.3

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.

(continued)



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR.

-----NOTE-----
 Pressurizer pressure limit does not apply during pressure transients due to:
 a. THERMAL POWER ramp > 5% RTP per minute; or
 b. THERMAL POWER step > 10% RTP.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is within limit specified in the COLR.	12 hours
SR 3.4.1.2 Verify RCS average temperature is within limit specified in the COLR.	12 hours
SR 3.4.1.3 -----NOTE----- Required to be performed within 7 days after \geq 95% RTP. ----- Verify RCS total flow rate is within the limit specified in the COLR.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be $\geq 540^\circ\text{F}$.

APPLICABILITY: MODE 1,
MODE 2 with $k_{off} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or both RCS loops not within limit.	A.1 Be in MODE 2 with $K_{off} < 1.0$.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.2.1 NOTE</p> <p>Only required if any 3.4.2.2¹ Verify RCS T_{avg} in each loop T_{avg} $< 547^\circ\text{F}$ and the low T_{avg} alarm is either inoperable or not reset $\geq 540^\circ\text{F}$.</p>	<p>Within 30 minutes prior to achieving criticality.</p>

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SR 3.4.2.2

NOTE

Only required if any RCS loop $T_{avg} < 547^{\circ}F$
and the low T_{avg} alarm is either inoperable
or not reset.

Verify RCS T_{avg} in each loop $\geq 540^{\circ}F$.

Once within
30 minutes and
every 30
minutes
thereafter

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Action A.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in MODE 1, 2, 3, or 4.</p>	<p>A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.</p>	<p>30 minutes 72 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < 500 psig.</p>	<p>6 hours 36 hours</p>

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Required Action C.2 shall be completed whenever this Condition is entered. -----</p> <p>Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.</p>	<p>C.1 Initiate action to restore parameter(s) to within limits.</p> <p><u>AND</u></p> <p>C.2 Determine RCS is acceptable for continued operation.</p>	<p>Immediately</p> <p>Prior to entering MODE 4</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.3.1 -----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -----</p> <p>Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.</p>	<p>30 minutes</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODE 1 > 8.5% RTP

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODE 1 > 8.5% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 1 \leq 8.5% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODES 1 ≤ 8.5% RTP, 2, and 3

LCO 3.4.5 Two RCS loops shall be OPERABLE and one loop shall be in operation.

-----NOTE-----
Both reactor coolant pumps may be de-energized in MODE 3 for ≤ 1 hour per 8 hour period provided:

- a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
-

APPLICABILITY: MODES 1 ≤ 8.5% RTP,
MODES 2 and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. -----	
	A.1 Verify SDM is within limits specified in the COLR. <u>AND</u> A.2 Restore inoperable RCS loop to OPERABLE status.	Once per 12 hours 72 hours

(continued)

ACTIONS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.5.3 Verify correct breaker alignment and indicated power are available to the required RCP that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and RHR pumps may be de-energized for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. No RCP shall be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR unless:
 - a. The secondary side water temperature of each steam generator (SG) is $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures; or
 - b. The pressurizer water volume is < 324 cubic feet (38% level).

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One RCS loop inoperable.</p> <p><u>AND</u></p> <p>Two RHR loops inoperable.</p>	<p>A.1 Initiate action to restore a second loop to OPERABLE status.</p>	<p>Immediately</p>
<p>B. One RHR loop inoperable.</p> <p><u>AND</u></p> <p>Two RCS loops inoperable.</p>	<p>-----NOTE----- Required Action B.1 is not applicable if all RCS and RHR loops are inoperable and Condition C is entered. -----</p> <p>B.1 Be in MODE 5.</p>	<p>24 hours</p>
<p>C. All RCS and RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RCS or RHR loop in operation.</p>	<p>C.1 Suspend all operations involving a reduction of RCS boron concentration.</p> <p><u>AND</u></p> <p>C.2 Initiate action to restore one loop to OPERABLE status and operation.</p>	<p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water level is $\geq 16\%$ for each required RCS loop.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

- a. One additional RHR loop shall be OPERABLE; or
- b. The secondary side water level of at least one steam generator (SG) shall be $\geq 16\%$.

-----NOTES-----

- 1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
- 2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
- 3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless:
 - a. The secondary side water temperature of each SG is $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures; or
 - b. The pressurizer water volume is < 324 cubic feet (38% level).
- 4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One RHR loop inoperable.</p> <p><u>AND</u></p> <p>Both SGs secondary side water levels not within limits.</p>	<p>A.1 Initiate action to restore a second RHR loop to OPERABLE status.</p> <p><u>OR</u></p> <p>A.2 Initiate action to restore required SG secondary side water levels to within limits.</p>	<p>Immediately</p> <p>Immediately</p>
<p>B. Two RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RHR loop in operation.</p> <p>B. Both RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RHR loop in operation.</p>	<p>B.1 Suspend all operations involving a reduction of RCS boron concentration.</p> <p><u>AND</u></p> <p>B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.</p>	<p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify one RHR loop is in operation.	12 hours

SURVEILLANCE	FREQUENCY
SR 3.4.7.2 Verify SG secondary side water level is \geq 16% in the required SG.	12 hours
(continued)	
SR 3.4.7.3 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

-----NOTES-----

1. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided:
 - a. No operations are permitted that would cause a reduction of the RCS boron concentration;
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature; and
 - c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>80</p> <p>B. Two RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RHR loop in operation.</p> <p>B. Both RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RHR loop in operation.</p>	<p>B.1 Suspend all operations involving reduction in RCS boron concentration.</p> <p><u>AND</u></p> <p>B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.</p>	<p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2 Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours
	<u>AND</u> A.2 Be in MODE 4.	12 hours
B. Pressurizer heaters capacity not within limits.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is \leq 87%.	12 hours
SR 3.4.9.2 Verify total capacity of the pressurizer heaters is \geq 100 Kw.	92 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2410 psig and ≤ 2545 psig.

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APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures greater than the
LTOP enable temperature specified in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Both pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.10.1 -----NOTE----- Required to be performed within 36 hours of entering MODE 4 from MODE 5 with all RCS cold leg temperatures greater than the LTOP enable temperature specified in the PTLR for the purpose of setting the pressurizer safety valves under ambient (hot) conditions only provided a preliminary cold setting was made prior to heatup. ----- Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 1\%$.</p>	<p>In accordance with the Inservice Testing Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. Separate entry into Condition A is allowed for each PORV.
 2. Separate entry into Condition C is allowed for each block valve.
 3. LCO 3.0.4 is not applicable.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both PORVs OPERABLE and not capable of being automatically controlled.	A.1 Close and maintain power to associated block valve.	1 hour
	<u>OR</u> A.2 Place associated PORV in manual control.	1 hour

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One PORV inoperable.</p>	<p>B.1 Close associated block valve.</p> <p><u>AND</u></p> <p>B.2 Remove power from associated block valve.</p> <p><u>AND</u></p> <p>B.3 Restore PORV to OPERABLE status.</p>	<p>1 hour</p> <p>1 hour</p> <p>72 hours</p>
<p>C. One or both block valves inoperable.</p> <p>C. 1 Place associated PORV(s) in manual control One block valve inoperable.</p> <p><u>AND</u></p> <p>C.2 Restore block valve(s) to OPERABLE status.</p>	<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p> <p>1 hour</p> <p>72 hours</p> <p>C.1 Place associated PORV in manual control.</p> <p><u>AND</u></p> <p>C.2 Restore block valve to OPERABLE status.</p>	<p>1 hour</p> <p>72 hours <u>7 days</u></p>

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. 1 Be in MODE 3 Both block valves inoperable.</p>	<p>E. (continued)</p>	<p>6 hours</p>
<p>AND</p>		<p>12-1 hour</p>
<p>D.2 Be in MODE 4.</p>	<p>12 hours D.1 Place associated PORVs in manual control.</p>	
	<p><u>AND</u> D.2 Restore at least one block valve to OPERABLE status.</p>	<p><u>72 hours</u></p>

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(continued)

<p>E. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p>	<p>E.1 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u></p>	
	<p>E.2 Be in MODE 4.</p>	<p>12 hours</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>☒ Two PORVs inoperable.</p>	<p>EF.1 Initiate action to restore one PORV to OPERABLE status.</p>	<p>Immediately</p>
	<p><u>AND</u></p>	
	<p>EF.2 Close associated block valves.</p>	<p>1 hour</p>
	<p><u>AND</u></p>	
	<p>EF.3 Remove power from associated block valves.</p>	<p>1 hour</p>
	<p><u>AND</u></p>	
	<p>EF.4 Be in MODE 3 with $T_{avg} < 500^{\circ}F$.</p>	<p>8 hours</p>

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.11.1 -----NOTE----- Not required to be performed with block valve closed per LCO 3.4.13. -----</p> <p>237</p> <p>Perform a complete cycle of each block valve.</p>	<p>92 days</p>
<p>SR 3.4.11.2 Perform a complete cycle of each PORV.</p>	<p>24 months</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with the Emergency Core Cooling System (ECCS) accumulators isolated and either a or b below.

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- a. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR and no safety injection (SI) pump capable of injecting into the RCS.
- b. The RCS depressurized and an RCS vent of ≥ 1.1 square inches and a maximum of one SI pump capable of injecting into the RCS.

-----NOTES-----

1. The PORVs and an RCS vent ≥ 1.1 square inches are not required to be OPERABLE during performance of the secondary side hydrostatic tests. However, no SI pump may be capable of injecting into the RCS during this test.
2. Accumulator/ECCS accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

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APPLICABILITY: MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or when the RHR system is in the RHR mode of operation,
MODE 5 when the SG primary system manway and pressurizer manway are closed and secured in position,
MODE 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>84</p> <p>A. NOTE Only applicable to LGO 3.4.12.a.</p> <p>83</p> <p>One or more SI pumps capable of injecting into the RCS.</p> <p>AND</p> <p>The PORVs provide the RCS vent path.</p>	<p>B. Immediately</p> <p>A.1 Initiate action to verify no SI pump is capable of injecting into the RCS.</p> <p>A.1 Initiate action to ensure no SI pump is capable of injecting into the RCS.</p> <p>verify →</p>	<p>Immediately</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>Immediately -----NOTE----- Only applicable to LCO 3.4.12.a.</p> <p>E. 12 hours</p> <p>12 hours D.1 Increase RCS cold leg temperature to greater than the LTOP enable temperature specified in the PTLR.</p> <p><u>OR</u></p> <p>D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.</p> <p>D. Required Action and associated Completion Time of Condition C not met. (continued)</p> <p>1 hour</p> <p>G.1 Isolate affected accumulator.</p>	<p>EB.1 Restore required PORV to OPERABLE status.</p>	<p>7 days</p>

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>FC. NOTE Only applicable to LCO 3.4.12.a. One required PORV inoperable in MODE 5 or MODE 6.</p>	<p>FC.1 Restore required PORV to OPERABLE status.</p>	<p>72 hours</p>
<p>D. Two or more SI pumps capable of injecting into the RGS. NOTE only applicable to LCO 3.4.12.b.</p>	<p>D.1 Initiate action to ensure a maximum of one SI pump is capable of injecting into the RGS.</p>	<p>(continued) Immediately</p>

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(continued)

~~G. Two required PORVs inoperable.~~

OR

~~Required Action and associated Completion Time of Condition A, D, E, or F not met.~~

OR

~~LTOP System inoperable for any reason other than Condition A, C, E, or F.~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>82 E. An EGCS accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR.</p>	<p>E.1. Isolate affected EGCS accumulator.</p>	<p>1 hour</p>
<p>81 F. Required Action and associated Completion Time of Condition E not met.</p>	<p>F.1. Increase RCS cold leg temperature to greater than the LTOP enable temperature specified in the PTLR.</p> <p>OR</p> <p>F.2. Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.</p>	<p>12 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G Two required PORVs inoperable for LCO 3.4.12.a.</p> <p>OR</p> <p>Required Action and associated Completion Time of Condition A, B, C, or F not met.</p> <p>OR</p> <p>LTOP System inoperable for any reason other than Condition A, B, C, or E.</p>	<p>G.1 Verify at least one charging pump is in the pull-stop position.</p> <p>AND</p> <p>G.2 Depressurize RCS and establish RCS vent of ≥ 1.1 square inches.</p>	<p>1 hour</p> <p>8 hours</p>

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.1 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.a. -----</p> <p>Verify no SI pump is capable of injecting into the RCS.</p>	<p>12 hours</p>
<p>SR 3.4.12.2 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. -----</p> <p>Verify a maximum of one SI pump is capable of injecting into the RCS.</p>	<p>12 hours</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
(continued)	
<p>SR 3.4.12.3 -----NOTE----- Only required to be performed when <u>ECCS</u> accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. ----- Verify each <u>ECCS</u> accumulator motor operated isolation valve is closed.</p>	<p>12 hours Once within 12 hours and every 12 hours thereafter</p>
<p>SR 3.4.12.4 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. ----- Verify RCS vent \geq 1.1 square inches open.</p>	<p>12 hours for unlocked open vent valve(s) <u>AND</u> 31 days for locked open vent valve(s)</p>
<p>SR 3.4.12.5 Verify PORV block valve is open for each required PORV.</p>	<p>72 hours</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.6 -----NOTE----- Required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR. -----</p> <p>Perform a COT on each required PORV, excluding actuation.</p>	<p>31 days</p>
<p>SR 3.4.12.7 -----NOTE----- Only required to be performed when <u>ECGS</u> accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. -----</p> <p>Verify power is removed from each <u>ECGS</u> accumulator motor operated isolation valve operator.</p>	<p>172</p> <p>Once within 12 hours and every 31 days thereafter</p>
<p>SR 3.4.12.8 Perform CHANNEL CALIBRATION for each required PORV actuation channel.</p>	<p>24 months</p>

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE; and
- d. 0.1 gpm total primary to secondary LEAKAGE through each steam generator (SG) when averaged over 24 hours.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>239</p> <p>B. Steam-Generator-Tube Surveillance Program Required Action and associated Completion Time not met.</p> <p>C. Required Action and associated Completion Time of Condition A or B not met.</p> <p>4 hours</p> <p>B.1 Determine steam generator tube integrity is acceptable for continued operation.</p> <p><u>OR</u></p> <p>RCS pressure boundary LEAKAGE exists.</p>	<p>CB.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>CB.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1 -----NOTE----- Only required to be performed during steady state operation. ----- Perform RCS water inventory balance.</p>	<p>Once during initial 12 hours of steady state operation <u>AND</u> 72 hours thereafter</p>
<p>SR 3.4.13.2 Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.</p>	<p>In accordance with the Steam Generator Tube Surveillance Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flowpaths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 or SR 3.4.14.2 and be in the reactor coolant pressure boundary or the high pressure portion of the system.</p> <p>-----</p>	<p>(continued)</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<p><u>AND</u></p> A.2 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<p><u>AND</u></p> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each SI cold leg injection line and each RHR RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.</p>	<p>24 months</p> <p><u>AND</u></p> <p>Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve</p>

(continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed until prior to entering MODE 2 from MODE 3. 2. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each SI hot leg injection line RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.</p>	<p>40 months</p> <p><u>AND</u></p> <p>Within 24 hours following valve actuation due to automatic or manual action, flow through the valve, or maintenance on the valve</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. One containment sump A monitor (level or pump actuation); and
- b. One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Required containment sump monitor inoperable.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>A.1.1 Perform SR 3.4.13.1.</p> <p style="text-align: center;"><u>OR</u></p> <p>A.1.2 Verify containment air cooler condensate collection system is OPERABLE.</p> <p style="text-align: center;"><u>AND</u></p> <p>A.2 Restore required containment sump monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>24 hours</p> <p>30 days</p>

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required containment atmosphere radioactivity monitor inoperable.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>B.1.1 Analyze grab samples of the containment atmosphere.</p> <p style="text-align: center;"><u>OR</u></p> <p>B.1.2 Perform SR 3.4.13.1.</p> <p style="text-align: center;"><u>AND</u></p> <p>B.2 Restore required containment atmosphere radioactivity monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>Once per 24 hours</p> <p>30 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required containment sump monitor inoperable.</p> <p><u>AND</u></p> <p>Particulate containment atmosphere radioactivity monitor inoperable.</p>	<p>-----NOTE----- LCO 3.0.4 is not applicable. -----</p> <p>C.1.1 Analyze grab samples of the containment atmosphere.</p> <p><u>OR</u></p> <p>C.1.2 Perform SR 3.4.13.1</p> <p><u>AND</u></p> <p>C.2.1 Restore required containment sump monitor to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2.2 Restore particulate containment atmosphere radioactivity monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>Once per 24 hours</p> <p>30 days</p> <p>30 days</p>
<p>D. Required Action and associated Completion Time of Conditions A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
<p>E. All required monitors inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitor.	92 days
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment sump monitor.	24 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	24 months

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~~24 months~~

~~SR 3.4.15.5~~

NOTE

~~Only required to be performed when complying with Required Action A.1.2 for LCO 3.4.15~~

~~Perform CHANNEL CALIBRATION of the required containment air cooler condensate system monitor.~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity not within limit.	-----NOTE----- LCO 3.0.4 is not applicable. -----	Once per 8 hours
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	7 days
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.	B.1 Be in MODE 3 with $T_{avg} < 500^\circ\text{F}$.	8 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)		
<p>THE FOLLOWING TEXT WAS MOVED G-C. THE PRECEDING TEXT WAS MOVED Gross specific activity not within limit.</p>	<p>1 Be in MODE 3 with $T_{avg} < 500^{\circ}F.$</p>	<p>8 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.16.1 Verify reactor coolant gross specific activity $\leq 100/E \mu Ci/gm.$</p>	<p>7 days</p>
<p>SR 3.4.16.2 -----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 1.0 \mu Ci/gm.$</p>	<p>14 days <u>AND</u> Between 2 and 10 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period</p>

SURVEILLANCE	FREQUENCY
(continued)	
<p>SR 3.4.16.3 -----NOTE----- Only required to be performed in MODE 1 within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.</p> <p>Determine \bar{E} from a reactor coolant sample.</p>	<p>184 days Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.</p> <p><u>AND</u></p> <p>Every 184 days thereafter</p>

(174)

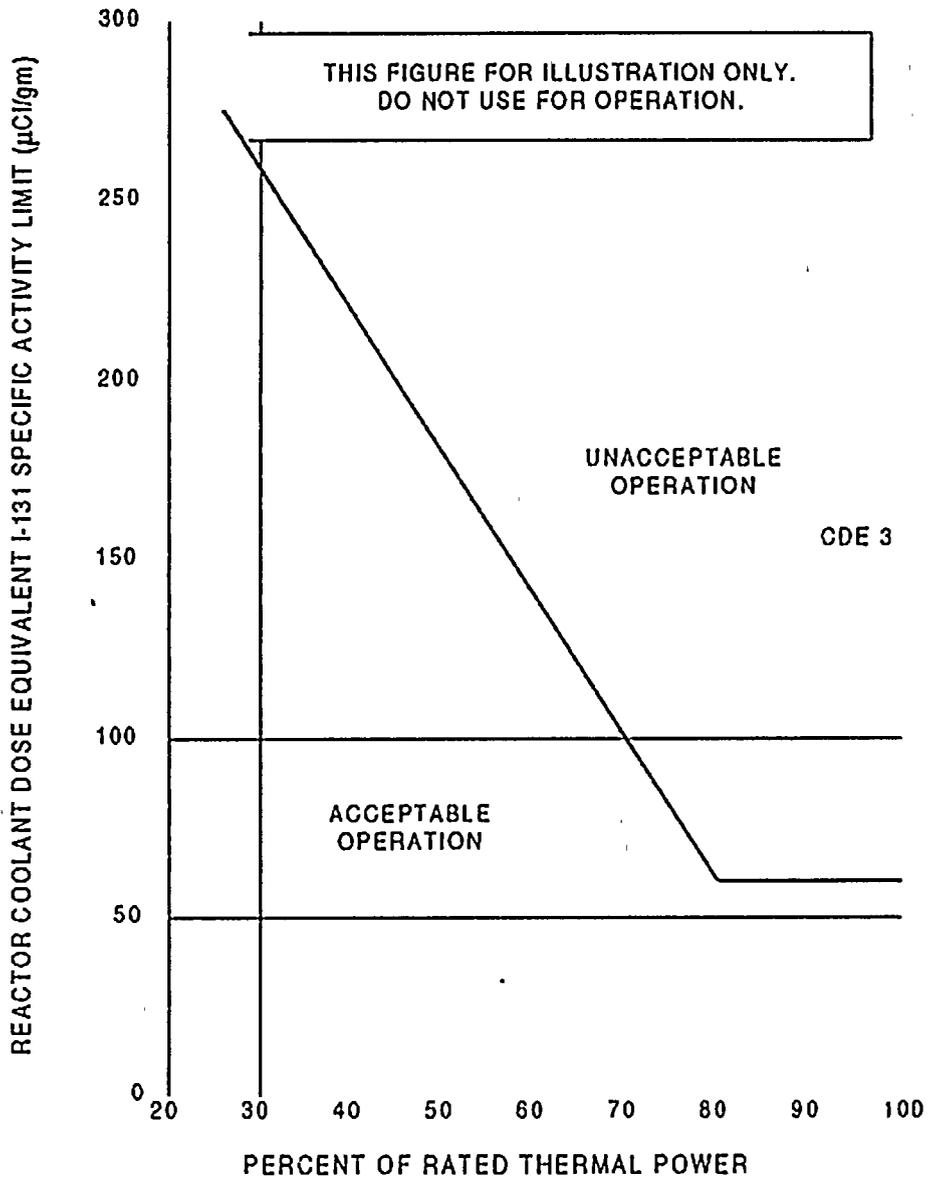


Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the departure from nucleate boiling (DNB) design criterion will be met for each of the transients analyzed.

The design method employed to meet the DNB design criterion for fuel assemblies is the Improved Thermal Design Procedure (ITDP). With the ITDP methodology, uncertainties in plant operating parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, ITDP design limit departure from nucleate boiling ratio (DNBR) values are determined in order to meet the DNB design criterion.

The ITDP design limit DNBR values are 1.34 and 1.33 for the typical and thimble cells, respectively, for fuel analyses with the WRB-2 correlation.

Additional DNBR margin is maintained by performing the safety analyses to DNBR limits higher than the design limit DNBR values. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. The safety analysis DNBR values are 1.52 and 1.51 for the typical and thimble cells, respectively.

(continued)

BASES

BACKGROUND
(continued)

For both the WRB-1 and WRB-2 correlations, the 95/95 DNBR correlation limit is 1.17. The W-3 DNB correlation is used where the primary DNBR correlations were developed based on mixing vane data and therefore are only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlations. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45. For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

The RCS pressure limit as specified in the COLR, is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit as specified in the COLR, is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate as specified in the COLR, normally remains constant during an operational fuel cycle with both pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNB design criterion. This is the acceptance limit for the RCS DNB parameters. Changes to the plant that could impact these parameters must be assessed for their impact on the DNB design criterion. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The limit for pressurizer pressure is based on a ± 30 psig instrument uncertainty. The accident analyses assume that nominal pressure is maintained at 2235 psig. By Reference 2, minor fluctuations are acceptable provided that the time averaged pressure is 2235 psig.

The RCS coolant average temperature limit is based on a $\pm 4^\circ\text{F}$ instrument uncertainty which includes a $\pm 1.5^\circ\text{F}$ deadband. It is assumed that nominal T_{avg} is maintained within $\pm 1.5^\circ\text{F}$ of 573.5°F . By Reference 2, minor fluctuations are acceptable provided that the time averaged temperature is within 1.5°F of nominal.

The limit for RCS flow rate is based on the nominal T_{avg} and SG plugging criteria limit. Additional margin of approximately 3% is then added for conservatism.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNB design criterion in the event of a DNB limited transient.

(continued)

BASES

LCO
(continued)

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNB design criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In MODE 2, an increased DNBR margin exists. In all other MODES, the power level is low enough that DNB is not a concern.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

(continued)

BASES

ACTIONS

A.1 (continued)

The 2 hour Completion Time for restoration of the parameters provides sufficient time to determine the cause for the off normal condition, to adjust plant parameters, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

Measurement of RCS total flow rate once every 24 months verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. This verification may be performed via a precision calorimetric heat balance or other accepted means.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. Verification of RCS flow rate on a shorter interval is not required since this parameter is not expected to vary during steady state operation as there are no RCS loop isolation valves or other installed devices which could significantly alter flow. Reduced performance of a reactor coolant pump (RCP) would be observable due to bus voltage and frequency changes, and installed alarms that would result in operator investigation.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the plant in the best condition for performing the SR. The Note states that the SR shall be performed within 7 days after reaching 95% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 95% RTP to obtain the stated RCS flow accuracies.

(continued)

BASES

REFERENCES

1. UFSAR, Chapter 15.
 2. NRC Memorandum from E.L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operations Inspection to Distribution; Subject: "Discussion of Licensed Power Level (AITS F14580H2)," dated August 22, 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

1c9
The first consideration is moderator temperature coefficient (MTC), LCO 3-1-43-13, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the RCS water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures greater than or equal to the HZP temperature of 547°F. The minimum temperature for criticality limitation provides a small band, 7°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{\text{off}} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1, and MODE 2 with $k_{\text{off}} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{\text{off}} \geq 1.0$) in these MODES.

(continued)

BASES

APPLICABILITY

169

(continued)

The special test exception of LCO ~~3.1.103.1.8~~, "MODE 2 PHYSICS TESTS Exceptions- MODE 2," permits PHYSICS TESTS to be performed at $\leq 5\%$ RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO. The need to perform the PHYSICS TESTS to ensure that the operating characteristics of the core are consistent with design predictions provides sufficient justification to allow a temporary decrease in the RCS minimum temperature for criticality limit.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $K_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period due to the proximity to MODE 2 conditions. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $K_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

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~~This SR verifies that RCS loop average temperature is required to be verified at or above 540°F every 30 minutes T_{avg} in MODE 2 with $k_{eff} \geq 1.0$ each loop is $\geq 540^\circ F$ within 30 minutes prior to achieving criticality. This ensures that the minimum temperature for criticality is being maintained just before criticality is reached.~~

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.4.2.2~~ (continued)
~~REQUIREMENTS~~

(76)

RCS loop average temperature is required to be verified at or above 540°F every 30 minutes in MODE 1, and in MODE 2 with $K_{eff} \geq 1.0$. The 30 minute time period is long enough to allow the operator to adjust temperatures or delay criticality so the LCO will not be violated, thereby providing assurance that the safety analyses are not violated.

REQUIREMENTS

~~SURVEILLANCE~~ ~~SR 3.4.2.1~~ (continued)

This SR is modified by a Note that only requires the SR to be performed if any RCS loop T_{avg} is $< 547^{\circ}\text{F}$ and the low T_{avg} alarm is either inoperable or not reset. The T_{avg} alarm provides operator indication of low RCS temperature without requiring independent verification while a $T_{avg} > 547^{\circ}\text{F}$ in both RCS loops is within the accident analysis assumptions. If the T_{avg} alarm is to be used for this SR, it should be calibrated consistent with industry standards.

(169)

This surveillance is replaced by SR ~~3-1-10-23~~ 18.2 during PHYSICS TESTING.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

(continued)



BASES

BACKGROUND
(continued)

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material has been established by periodically removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves have been adjusted based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be $\geq 40^\circ\text{F}$ above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

(continued)

BASES

BACKGROUND
(continued)

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE
SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and result in nonductile failure of the RCPB which is an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

- a. The limit curves for heatup, cooldown, and ISLH testing; and
- b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

(continued)



BASES

LCO
(continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and
- c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

(continued)

BASES

APPLICABILITY
(continued)

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

(continued)

BASES

ACTIONS

A.1 (continued)

Condition A is modified by a Note stating that Required Action A.2 shall be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event which is best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished quickly in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1 (continued)

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994.
 2. 10 CFR 50, Appendix G.
 3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
 4. ASTM E 185-82, July 1982.
 5. 10 CFR 50, Appendix H.
 6. Regulatory Guide 1.99, Revision 2, May 1988.
 7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODE 1 > 8.5% RTP

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;
- b. Improving the neutron economy by acting as a reflector;
- c. Carrying the soluble neutron poison, boric acid; and
- d. Providing a second barrier against fission product release to the environment.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE
SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds all possible instrumentation errors (Ref. 2). The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with both RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. Adequate heat transfer between the reactor coolant and the secondary side is ensured by maintaining \geq 16% SG level in accordance with LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," which provides sufficient water inventory to cover the SG tubes.

RCS Loops - MODE 1 > 8.5% RTP satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required to be in operation at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 1 > 8.5% RTP, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, both RCS loops are required to be OPERABLE and in operation in this MODE to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower MODES as indicated by the LCOs for MODES $1 \leq 8.5\%$ RTP, 2, 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODES $1 \leq 8.5\%$ RTP, 2, AND 3";

LCO 3.4.6, "RCS Loops - MODE 4";

LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

~~LCO 3.9.3, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft" (MODE 6);~~
~~and~~

LCO-3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft" (MODE 6);
and

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft" (MODE 6).

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(continued)

BASES

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 1 < 8.5% RTP. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

(continued)

BASES

~~The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging safety systems.~~ ACTIONS A.1 (continued)

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The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

REFERENCES

1. UFSAR, Chapter 15.
 2. UFSAR, Section 15.0.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODES 1 \leq 8.5% RTP, 2, AND 3

BASES

BACKGROUND

In MODE 1 \leq 8.5% RTP, and in MODE 2 and 3, the primary function of the RCS is the removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant. The secondary functions of the RCS include:

- (149)
- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission (MODE 1 \leq 8.5% RTP and MODE 2 only);
 - b. Improving the neutron economy by acting as a reflector (MODE 1 \leq 8.5% RTP and MODE 2 only);
 - c. Carrying the soluble neutron poison, boric acid; and
 - d. Providing a second barrier against fission product release to the environment.

The reactor coolant is circulated through two RCS loops, connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 1 \leq 8.5% RTP and MODE 2, the RCPs are used to provide forced circulation of the reactor coolant to ensure mixing of the coolant for proper boration and chemistry control and to remove the limited amount of reactor heat. In MODE 3, the RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 1 \leq 8.5% RTP, 2, and 3 reactor and decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). In MODE 1 \leq 8.5% RTP, and in MODES 2 and 3, these analyses include evaluation of main steam line breaks and uncontrolled rod withdrawal from a subcritical condition. The most limiting accident with respect to DNB limits for MODES 1 \leq 8.5% RTP, 2, and 3 is a main steam line break. This is due to the potential for recriticality and because of the high hot channel factors that may exist if the most reactive control rod is stuck in its fully withdrawn position.

A main steam line break has been analyzed for both the case with one and two RCS loops in operation at hot zero power (HZP) conditions with acceptable results (Ref. 1). However, with only one RCS loop in operation and offsite power available, additional shutdown margin is required since the reduced flow produces an adverse effect on DNB limits.

The startup of an inactive reactor coolant pump (RCP) up to 8.5% RTP has been evaluated and found to result in only limited power and temperature excursions that are bounded by a main steam line break with only one RCS Loop in operation (Refs. 2 and 3).

Analyses have also been performed which demonstrate that reactor heat greater than 5% RTP can be removed by natural circulation alone (Ref. 4).

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. RCS Loops - MODES 1 \leq 8.5 % RTP, 2, and 3 satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

The purpose of this LCO is to require that both RCS loops be OPERABLE. Only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS up to 8.5% RTP. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met. Requiring one RCS loop in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

The Note permits all RCPs to be de-energized for \leq 1 hour per 8 hour period in MODE 3. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test was satisfactorily performed during the initial startup testing program (Ref. 5). If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

The no flow test may be performed in MODE 3, 4, or 5. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

(continued)

BASES

LCO
(continued)

- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and able to provide forced flow if required.

APPLICABILITY

In MODES 1 \leq 8.5% RTP, 2, and 3, this LCO ensures forced circulation of the reactor coolant to remove reactor and decay heat from the core and to provide proper boron mixing.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO ~~3.9.3, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level \geq 23 Ft" (MODE 6);~~
~~and~~
LCO-3.9.4, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level \leq 23 Ft" (MODE 6);
and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level \leq 23 Ft" (MODE 6).

ACTIONS

A.1 and A.2

If one RCS loop is inoperable, redundancy for heat removal is lost. The Required Actions are to verify that the SDM is within limits specified in the COLR. This action is required to ensure that adequate SDM exists in the event of a main steam line break with only one RCS loop in operation. The 12 hour Frequency considers the time required to obtain

(continued)



BASES

RCS boron concentration samples and the low probability of a main steam line break during this time period.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The inoperable RCS loop must be restored to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the reactor and decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one RCS loop is inoperable. This allowance is provided because a single RCS loop can provide the required cooling to remove reactor and decay heat consistent with safety analysis assumptions.

(7)

B.1

If restoration of the inoperable loop is not possible within 72 hours, the plant must be brought to MODE 4. In MODE 4, the plant may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1, C.2, and C.3

If two RCS loops are inoperable, or no RCS loop is in operation, except during conditions permitted by the Note in the LCO section, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that each required RCS loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of the control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is \geq 16% for two RCS loops. If the SG secondary side narrow range water level is $<$ 16%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of reactor or decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.5.3

Verification that the required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR Section 15.1.5.
 2. UFSAR Section 15.4.3.
 3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-9, Startup of an Inactive Loop, R. E. Ginna," dated August 26, 1981.
 4. UFSAR Sections 14.6.1.5.6 and 15.2.5.2.
 5. UFSAR Section 14.6.1.5.5.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the cladded fuel. The SGs or the RHR heat exchangers provide the heat sink. The RCPs and the RHR pumps circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCS or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCS or one RHR loop for decay heat removal and transport. The flow provided by one RCS loop or one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 have been identified in the NRC Policy Statement as important contributors to risk reduction.

(continued)

BASES

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

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Note 1 permits all RCPs ~~or~~ and RHR pumps to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program was the validation of rod drop times during cold conditions, both with and without flow (Ref. 1). If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

(continued)

BASES

LCO
(continued)

Note 2 requires that the pressurizer water volume be < 324 cubic feet (38% level), or that the secondary side water temperature of each SG be $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed condition.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2. RCPs are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. An OPERABLE RHR loop may be isolated from the RCS provided that the loop can be placed into service from the control room. RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

(continued)

BASES

APPLICABILITY
(continued)

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
LCO 3.4.5, "RCS Loops - MODES 1 ≤ 8.5% RTP, 2, AND 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
~~LCO 3.9.3, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);
and~~
LCO-3.9.4, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level < 23 Ft" (MODE 6).

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ACTIONS

A.1

If one RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. If no RHR is available, the plant cannot enter a reduced MODE since no long term means of decay heat removal would be available. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one RHR loop is inoperable and both RCS loops are inoperable, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

No change

~~ACTIONS - If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. (continued)~~

~~If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost~~

(continued)

BASES

and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 ($\leq 200^{\circ}\text{F}$) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

(continued)

BASES

No change

~~Required Action-ACTIONS~~

~~B.1 (continued)~~

~~Required Action B.1~~ is modified by a Note stating that only the Required Actions of Condition C are entered if all RCS and RHR loops are inoperable. With all RCS and RHR loops inoperable, MODE 5 cannot be entered and Required Actions C.1 and C.2 are the appropriate remedial actions.

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

(continued)

BASES

SURVEILLANCE

No change

(continued)

SR 3.4.6.2

REQUIREMENTS

This SR requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq 16\%$. If the SG secondary side narrow range water level is $< 16\%$, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump that is not in operation. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 14.6.1.2.6.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat either to the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is normally circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining one SG with a secondary side water level at or above 16% to provide an alternate method for decay heat removal.

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(continued)

BASES

APPLICABLE
SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or one SG with a secondary side water level $\geq 16\%$. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is one SG with a secondary side water level $\geq 16\%$. Should the operating RHR loop fail, the SG could be used to remove the decay heat.

Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program was the validation of rod drop times during cold conditions, both with and without flow (Ref. 1). If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

(continued)

BASES

LCO
(continued)

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period \leq 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the pressurizer water volume be $<$ 324 cubic feet (38% level), or that the secondary side water temperature of each SG be \leq 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature are to prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed Condition.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops. A planned heatup is a scheduled transition to MODE 4 within a defined time period.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it is OPERABLE in accordance with the Steam Generator Tube Surveillance Program, with the minimum water level specified in SR 3.4.7.2.

(continued)

BASES

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The RCS loops are considered filled until the isolation valves are opened to facilitate draining of the RCS. The loops are also considered filled following the completion of filling and venting the RCS. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be $\geq 16\%$.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 $> 8.5\%$ RTP";
LCO 3.4.5, "RCS Loops - MODES 1 $\leq 8.5\%$ RTP, 2, AND 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
~~LCO 3.9.3, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level < 23 Ft" (MODE 6).~~

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ACTIONS

A.1 and A.2

If one RHR loop is inoperable and both SGs have secondary side water levels $< 16\%$, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore at least one SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until an RHR loop is restored to OPERABLE status or SG secondary side water level is restored.

(continued)

BASES

ACTIONS
(continued)

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B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1, ~~2~~, and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. Use of control board indication for these parameters is an acceptable verification. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

This SR requires verification of SG OPERABILITY. Verifying that at least one SG is OPERABLE by ensuring its secondary side narrow range water level is $\geq 16\%$ ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby RHR pump. If secondary side water level is $\geq 16\%$ in at least one SG, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 14.6.1.2.6
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE
SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation to transfer heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one operating RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

(continued)



BASES

LCO
(continued)

Note 1 permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and requires that the following conditions be met:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation;
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and
- c. No draining operations are permitted that would further reduce the RCS water volume and possibly cause a more rapid heatup of the remaining RCS inventory.

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. The RCS loops are considered not filled from the time period beginning with the opening of isolation valves and draining of the RCS and ending with the completion of filling and venting the RCS.

(continued)



BASES

APPLICABILITY
(continued)

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODE 1 > 8.5% RTP";
LCO 3.4.5, "RCS Loops - MODES 1 ≤ 8.5% RTP, 2, AND 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
~~LCO 3.9.3, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);~~
~~and~~
LCO-3.9.4, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level ≥ 23 Ft" (MODE 6);
and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant
Circulation - Water Level < 23 Ft" (MODE 6).

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ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. The action to restore must continue until the second RHR loop is restored to OPERABLE status.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

(continued)

BASES

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level and the required heater capacity. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of this LCO is to ensure that a steam bubble exists in the pressurizer prior to, and during, power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensable gases are typically present in the RCS and can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control. These noncondensable gases can be ignored if the steam bubble is present.

(continued)



BASES

BACKGROUND
(continued)

This LCO also ensures that adequate heater capacity is available in the pressurizer to support natural circulation following an extended loss of offsite power. Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. These heaters are divided into two groups, a control/variable group and a backup group. The control/variable group is normally used during power operation since these heaters have inverse proportional control with respect to the pressurizer pressure. The backup group is either fully on or off with setpoints that are below those for the control/variable group. Both groups of heaters receive power from the Engineered Safety Feature (ESF) 480 V buses, however, the heaters are shed following a loss of offsite power or safety injection signal. The heaters can be manually loaded onto the diesel generators if required.

A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained during natural circulation. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat. Unless adequate heater capacity is available, the required subcooling margin in the primary system cannot be maintained. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat. Maintaining necessary subcooled margin during normal power operation is controlled by meeting the requirements for pressurizer level and LCO 3.4.1, "RCS Pressure, Temperature and Flow Departure From Nucleate Boiling (DNB) Limits."

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(continued)

BASES

APPLICABLE
SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting with respect to pressurizer parameters. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

The maximum pressurizer water level limit ensures that a steam bubble exists and satisfies Criterion 2 of the NRC Policy Statement.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation, however, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO. The pressurizer heaters are assumed to be available within one hour following the loss of offsite power and initiation of natural circulation (Ref. 3).

LCO

The LCO establishes the minimum conditions required to ensure that a steam bubble exists within the pressurizer and that sufficient heater capacity is available to support an extended loss of offsite power event. For the pressurizer to be considered OPERABLE, the limits established in the SRs for water level and heater capacity must be met and the heaters must be capable of being powered from an emergency power source within one hour.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3 to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

(continued)

BASES

APPLICABILITY
(continued)

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply (Ref. 4). In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

If the pressurizer water level is > 650 cubic feet, which is equivalent to 87%, the ability to maintain a steam bubble may no longer exist. The steam bubble is necessary to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit is the same as the Pressurizer High Level Trip.

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the plant must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the pressurizer heaters capacity is < 100 KW, the ability to maintain RCS pressure to support natural circulation may no longer exist. By maintaining RCS pressure control, a margin to subcooling is provided. The value of 100 KW is based on the amount needed to support natural circulation after accounting for heat losses through the pressurizer insulation during an extended loss of offsite power event.

If the capacity of the pressurizer heaters is not within the limit, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

This SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power required. This may be done by testing the power supply output by verifying the electrical load on Buses 14 and 16 with the respective heater groups on and off. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

(continued)

BASES

REFERENCES

1. UFSAR, Chapter 15.
 2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
 3. Letter from B. L. King, Westinghouse Electric Corporation, to R. C. Mecredy, RG&E, Subject: "Ability to Maintain Subcooled Conditions During an Extended Loss of Offsite Power," dated September 26, 1979.
 4. Letter from D. M. Crutchfield, NRC, to L. D. White, Jr. RG&E, Subject: "Lessons Learned Category 'A' Evaluation," dated July 7, 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 288,000 lbm/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5 and in MODE 6 with reactor vessel head on; however, in MODE 4, with either RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR, and MODE 5 and MODE 6 with the reactor vessel head on and the SG primary system manway and pressurizer manway closed and secured in position, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the $\pm 1\%$ tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

(continued)

BASES

BACKGROUND
(continued)

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the locked rotor accident which remains below 120% of the design pressure consistent with the original maximum transient pressure limit for the RCS (Refs. 2, 3 and 4). The consequences of exceeding the American Society of Mechanical Engineers (ASME) and USAS Section B31.1 pressure limits (Refs. 1 and 4) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE
SAFETY ANALYSES

All accident and safety analyses in the UFSAR (Ref. 5) that require safety valve actuation assume operation of both pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 6) is also based on operation of both safety valves. Accidents that could result in overpressurization if not properly terminated include:

- a. Uncontrolled rod withdrawal from full power;
- b. Loss of reactor coolant flow;
- c. Loss of external electrical load (including the complete loss of steam flow to the turbine);
- d. Loss of normal feedwater;
- e. Loss of all AC power to station auxiliaries; and
- f. Locked rotor.

Detailed analyses of the above transients are contained in Reference 5. Safety valve actuation is required in events c, d, e, and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits following testing are based on the $\pm 1\%$ tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The OPERABILITY limits of + 2.4%, - 3% are based on the analyzed events. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure for all transients except locked rotor accidents which has an allowed limit of 120% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when either RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head detensioned or the SG primary system manway or the pressurizer manway open.

(continued)

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with either RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperature at or below the LTOP enable temperature specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by both pressurizer safety valves.

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 7), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

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The pressurizer safety valve setpoint is + 2.4%, - 3% for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1 (continued)

This SR is modified by a Note that allows entry into MODES 3 and 4 without having performed the SR for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition until completion of the surveillance.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
 2. UFSAR, Section 15.3.2.
 3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic XV-1, XV-2, XV-3, XV-4, XV-5, XV-6, XV-7, XV-8, XV-10, XV-12, XV-14, XV-15, and XV-17, Design Basis Events, Accidents, and Transients (R.E. Ginna)," dated September 4, 1981.
 4. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967 edition.
 5. UFSAR, Chapter 15.
 6. WCAP-7769, "Topical Report, Overpressure Protection for Westinghouse Pressurized Water Reactors," Rev. 1, June 1972.
 7. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs (430 and 431C) are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Motor operated block valves (515 and 516), which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater and auxiliary feedwater. The PORVs are also used to mitigate the effects of an anticipated transient without scram (ATWS) event which is also not within the design basis.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The two PORVs (in manual operation only) and their associated block valves are powered from two separate safety trains.

(continued)

BASES

BACKGROUND
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The plant has two PORVs, each having a relief capacity of 179,000 lb/hr at 2335 psig. The PORVs are normally opened by using instrument air which is supplied through separate solenoid operated valves (8620A and 8620B). The safety related source of motive air is from two separate nitrogen accumulators that are normally isolated from the PORVs by solenoid operated valves 8619A and 8619B; however, solenoid operated valves 8620A and 8620B must be in the vent position to close the PORVs regardless of which motive air source is used.

The functional design of the PORVs is based on maintaining pressure below the pressurizer high pressure reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE
SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also used in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the pressurizer high pressure trip and pressurizer safety valve setpoints; thus the DNBR calculation is more conservative assuming the same initial RCS temperature since the pressurizer pressure is limited. Events that assume this condition include a loss of external electrical load and other transients which result in a decrease in heat removal by the secondary system (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation by the nitrogen accumulators to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV is required to be OPERABLE to mitigate the effects associated with an SGTR and its block valve must be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to automatically open with a subsequent failure to close. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high.

The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves. Therefore, the LCO is applicable in MODES 1, 2, and 3.

The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

(continued)

BASES

ACTIONS

Note 1 has been added to clarify that both pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis) for Condition A. Note 2 has been added to clarify that both block valves are treated as separate entities, each with separate Completion Times, for Condition C. The exception for LCO 3.0.4, Note 3, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES due to LTOP considerations.

A.1 and A.2

With the PORVs OPERABLE and not capable of being automatically controlled, either the PORVs must be restored or the flow path isolated within 1 hour. Although a PORV may not be capable of being automatically controlled, it may be able to be manually opened and closed, and therefore, able to perform its function. A PORV is considered not capable of being automatically controlled for any problem which prevents the PORV from automatically closing once it has automatically opened. This may be due to

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Not capable of automatic control

instrumentation problems, but does not include problems which only prevent the PORV from automatically opening (e.g., loss of instrument air to the PORV), or which prevent the PORV from both automatically opening and automatically closing. For these reasons, the block valve may either be closed to isolate the flowpaths but the Action requires power be maintained to the valve or isolated by placing the PORV control switch in the closed position. However, if the block valve is closed to isolate the flowpath may also, the Action requires power be isolated by placing the PORV control switch to manual maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2). Seat leakage problems are controlled by LCO 3.4.13, "RCS Operational LEAKAGE."

... also
due to
problems

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(11)

(continued)

21
BASES

~~Quick access to the PORV for pressure control can be made when power remains on the closed block valve~~ ACTIONS A.1 and A.2 (continued)

No change

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

ACTIONS—B.1, B.2, and B.3

~~(continued)~~

If one PORV is not capable of being manually cycled, it is inoperable and must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. PORV inoperability includes (but is not limited to) the inability of the solenoid operated isolation valve from the nitrogen accumulator to open or the solenoid operated isolation valve from instrument air to vent. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is a second PORV that is OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. E

(169)

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

(117)

If one ~~or both block valves are~~ block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. Manual control is accomplished by placing the PORV control board switch in the closed position. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because the PORV is not capable of automatically opening and the small potential for an SGTR or other event requiring Manual operation, the operator is permitted a Completion Time of ~~72 hours~~ 7 days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is limited to ~~72 hours~~ 7 days since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If the block valve is restored within the Completion Time of ~~72 hours~~ 7 days, the PORV will again be capable of automatically responding to an overpressure event, and the block valves capable of isolating a stuck open PORV which may result from the overpressure event. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition DE.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

(17)

If the Required Action of Condition A, B, or C is not met both block valves are inoperable, then the plant must be brought to a MODE it is necessary to either restore at least one block valve to OPERABLE status within the Completion Time of 1 hour or place the PORVs in which the LCO does not apply manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valves cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORVs in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. Manual control is accomplished by placing the PORV control board switch in the closed position. The Completion Time of 1 hour is reasonable based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because the PORV is not capable of automatically opening and the small potential for an SGTR or other event requiring Manual operation, the operator is permitted a Completion Time of 72 hours to restore at least one inoperable block valve to OPERABLE status. The time allowed to restore one block valve is limited to 72 hours since the PORVs are not capable of automatically mitigating an overpressure event when placed in manual control. If at least one block valve is restored within the Completion Time of 72 hours, at least one PORV will again be capable of automatically responding to an overpressure event, and the associated block valve capable of isolating a stuck open PORV which may result from the overpressure event. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition E.

(continued)

BASES

ACTIONS E.1 and E.2
(continued)

(117)

If the Required Action of Condition A, B, C, or D is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

(117)

E.1, E.2, E.3, and E.F.1, F.2, F.3, and F.4

If both PORVs are not capable of being manually cycled, they are inoperable and it is necessary to initiate action to restore one PORV to OPERABLE status immediately since no relief valve is available to mitigate the effects associated with an SGTR. Therefore, operators must either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

(continued)

BASES

ACTIONS

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EP.1, EP.2, EP.3, and EP.4 (continued)

If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two PORVs inoperable. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE which does not require manual PORV operation. To achieve this status, the plant must be brought to MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 8 hours. In MODE 3 with the RCS average temperature $< 500^{\circ}\text{F}$, the saturation pressure of the reactor coolant is below the setpoint of the main steam safety valves. Since the RWST contains a larger volume of water than the secondary side of an SG, the leak through the ruptured tube will stop after the SG is filled to capacity. Therefore, an SGTR can be mitigated under these conditions without any release of radioactive fluid through the main steam safety valves. Entering a lower MODE is not desirable with both PORVs inoperable and not capable of being manually cycled since the PORVs are also required for low temperature overpressure protection. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 2). If the block valve is

and is

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~~closed to isolate a PORV that is capable of being manually cycled, but not leaking in excess of the limits of LCO~~

using
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~~3.4.13, "RCS Operational LEAKAGE," the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed~~

the
can

to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days.

Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed per LCO 3.4.13. This prevents the need to open the block valve when the associated PORV is leaking > 10 gpm creating the potential for a plant transient.

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SR 3.4.11.2

This SR requires a complete cycle of each PORV using the nitrogen accumulators. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. UFSAR, Section 15.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The LTOP system also protects the RHR system from overpressurization during the RHR mode of operation. The PTLR provides the maximum allowable actuation logic setpoints for the pressurizer power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

(continued)

BASES

BACKGROUND
(continued)

This LCO provides RCS overpressure protection by restricting coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires isolating the Emergency Core Cooling System (ECCS) accumulators and rendering all safety injection (SI) pumps incapable of RCS injection when the PORVs provide the RCS vent path and rendering a minimum of two SI pumps incapable of RCS injection when the RCS is depressurized with an RCS vent ≥ 1.1 square inches. The pressure relief capacity requires either two redundant PORVs or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

By restricting coolant input capability, the ability to provide core coolant addition is minimized. The LCO does not require the makeup control system to be deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If the conditions require the use of SI for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The two redundant PORVs or a depressurized RCS with an open RCS vent is also sufficient to protect the RHR system during the RHR mode of operation for events which cause an increase in system pressure.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure exceeds the limit selected to prevent a condition that is not within the acceptable region provided in the PTLR. The PORVs are opened by coincident actuation of two-of-three RCS pressure channels. The PTLR presents the PORV setpoint for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and then reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

(continued)

BASES

BACKGROUND
(continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV's internals or blocking it open, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent path. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE
SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits for all Design Basis Accidents. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding the LTOP enable temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At or below the LTOP enable temperature specified in the PTLR, overpressure prevention falls to two OPERABLE PORVs or ~~⊕~~ a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability.

Requirements

overpressure protection systems

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as a result of

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases ~~due to~~ neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 3 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

- a. Inadvertent safety injection (SI); or
- b. Charging/letdown flow mismatch.

Heat Input Type Transients

- a. Inadvertent actuation of pressurizer heaters;
- b. Loss of RHR cooling; or
- c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

Analyses have determined that the mass input transients are the bounding case for overpressurization of the RCS (Ref. 3). The two categories of mass input transients were analyzed with respect to utilizing a single PORV or an RCS vent ≥ 1.1 square inches as overpressure protection. The inadvertent actuation of a single SI pump provides a larger mass addition to the RCS than isolation of letdown with all three charging pumps operating. A single PORV was determined to be incapable of mitigating the overpressure transient resulting from actuation of a SI pump, but is capable of mitigating the charging/letdown mismatch transient. An RCS vent ≥ 1.1 square inches can mitigate both the inadvertent SI and charging/letdown flow mismatch transients.

Therefore, the following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. Rendering all SI pumps incapable of injection into the RCS when the PORVs provide the RCS vent path and rendering all but one SI pump incapable of injection into the RCS when the RCS is depressurized with an RCS vent of ≥ 1.1 square inches;
- 82 b. Deactivating the EGCS accumulator discharge motor operated isolation valves in their closed positions; and
- 109 c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop or pressurizer level $\leq 38\%$. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

83 The Reference 3 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits with the maximum allowed coolant input capability. Since neither one PORV nor the RCS vent can handle the pressure transient produced from EGCS accumulator injection when RCS temperature is low, the LCO also requires the EGCS accumulators isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

87 The isolated EGCS accumulators must have their discharge valves closed and the valve power supply removed. The analyses show the effect of EGCS accumulator discharge is over a narrower RCS temperature range (200°F and below) than that of the LCO. Fracture mechanics analyses established the temperature of LTOP Applicability at the LTOP enable temperature specified in the PTLR.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having procedures to manually establish makeup capability.

The events which potentially overpressurize the RHR system during the RHR mode of operation are included within the mass and heat input transients analyzed for LTOP conditions. Therefore, an OPERABLE LTOP System ensures that the RHR system will not be overpressurized during the RHR mode of operation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient for the PORVs of a charging/letdown flow mismatch. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met and that the RHR system will not be overpressurized.

The PORV setpoints in the PTLR are updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 1.1 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, which maintains RCS pressure less than the maximum pressure on the P/T limit curve. The limiting transient for this LTOP configuration is an SI actuation with one SI pump OPERABLE.

An RCS vent ≥ 1.1 square inches with the RCS depressurized also prevents overpressurization of the RHR system.

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BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of the NRC Policy Statement.

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

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To limit the coolant input capability, the LCO requires the EGCS accumulators to be isolated. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," defines SI actuation OPERABILITY for the LTOP MODE 4 small break LOCA.

The elements of the LCO that provide low temperature overpressure mitigation are:

- a. Two OPERABLE PORVs and no SI pump capable of injecting into the RCS.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the valve and its control circuits.

- b. A depressurized RCS and an RCS vent and a maximum of one SI pump capable of injecting into the RCS.

An RCS vent is OPERABLE when open with an area of ≥ 1.1 square inches.

(continued)

BASES

LCO
(continued)

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The LCO is modified by two Notes. The first Note allows performance of the secondary side hydrostatic tests without the PORVs and RCS vent OPERABLE; however no SI pump may be capable of injecting into the RCS during this test. This exclusion is necessary since a pressure differential of ≤ 800 psid is maintained between the primary and secondary sides during the test. This restricted pressure differential limits the stresses placed on the SG which can cause cladding in the primary channel to separate from the base metal and result in the need for difficult repairs in a high radiation area. To maintain this pressure differential limit, RCS pressure must be increased above the PORV setpoint for LTOP conditions. The test cannot be performed above the LTOP enable temperature since the steam lines may not be able to accommodate the associated thermal expansion if they are heated. Therefore, all three SI pumps must be incapable of injecting into the RCS during these secondary side hydrostatic tests (Ref. 6).

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The second Note only requires an ECCS accumulator to be isolated when the accumulator pressure is greater than or equal to the maximum pressure for the existing RCS cold leg temperature allowed in the PTLR. Accumulator pressure below this limit will not overpressurize the RCS beyond analyzed conditions. The accumulator is isolated when the discharge motor operated valve is closed and its associated power supply is removed.

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BASES

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR or the RHR system is in the RHR operating mode, in MODE 5 when the SG primary system manway and pressurizer manway are closed and secured in position, and in MODE 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enable temperature specified in the PTLR. When the reactor vessel head is off or the SG primary system manway or pressurizer manway are open, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above the LTOP enable temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

A.1

With one or more SI pumps capable of injecting into the RCS and the PORVs provide the RCS vent path, RCS overpressurization is possible.

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■ To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

↑
taking action
to remove

↑
potential

(continued)

BASES

ACTIONS

(84)

Condition A is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

(continued)

BASES

ACTIONS B.1 (continued)

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

84

Condition B is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

C.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PORVs inoperable in MODE 5 with the SG primary system manway and pressurizer manway closed and secured in position, or in MODE 6 with the head on and the SG primary system manway and pressurizer manway closed and secured in position, the PORV must be restored to OPERABLE status in 72 hours. Restoring the PORV to OPERABLE status provides required redundancy.

The Completion Time of 72 hours to restore the PORV to OPERABLE status represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one PORV to protect against overpressure events. The Completion Time is also consistent with the time allowed for restoration of one train of ECCS, LCO 3.5.2 "ECCS Operating".

Condition C is modified by a Note which states that this condition is only applicable to LCO 3.4.12.a (i.e., when the PORVs provide the RCS vent path).

(continued)

Condition D is redefined by a Note which states that this condition is only applicable to LCO 2.4.2b C1e, when there is a RCS vent path ≥ 1.1 square inches

BASES

ACTIONS
(continued)

D.1

With two or more SI pumps capable of injecting into the RCS and the RCS is depressurized with an RCS vent of ≥ 1.1 square inches, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

potential

to take action to remove

81

D.1, D.1, and D.2

82

An unisolated ECCS accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

83

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to greater than the LTOP enable temperature specified in the PTLR, a maximum accumulator pressure of 800 psig (relief valve setpoint) cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

84

E.1

In MODE 4 when any RCS cold leg temperature is less than or equal to the LTOP enable temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

(continued)

BASES

ACTIONS ~~E.1 (continued)~~

~~The Completion Time considers that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.~~

F.1

~~The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PORVs inoperable in MODE 5 with the SG primary system manway and pressurizer manway closed and secured in position, or in MODE 6 with the head on and the SG primary system manway and pressurizer manway closed and secured in position, the PORV must be restored to OPERABLE status in 72 hours. Restoring the PORV to OPERABLE status provides required redundancy.~~

~~The Completion Time of 72 hours to restore the PORV to OPERABLE status represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one PORV to protect against overpressure events. The Completion Time is also consistent with the time allowed for restoration of one train of ECCS, LCO 3.5.2 "ECCS Operating".~~

G.1 and G.2

At least one charging pump must be in the pull-stop position within 1 hour and the RCS must be depressurized and a vent must be established within 8 hours when:

- a. Both required PORVs are inoperable; or
- b. A Required Action and associated Completion Time of Condition A, ~~BB~~, ~~CC~~, or F is not met; or

(continued)

BASES

~~ACTIONS G. — The LTOP System is inoperable for any reason other than Condition A, C, E, or F1 and G.~~

81

(continued)

BASES

ACTIONS — G2 (continued)

(24)

- c. ~~1 and 6~~ The LTOP System is inoperable for any reason other than Condition A, B, C, or E.

~~2~~ (continued)

The Completion Time of one hour to restrict the coolant input capability to the RCS considers the relatively low probability of an overpressure event during this time period and provides the operator time to render a charging pump incapable of injecting by placing it in the pull-stop position. Only one disabling device is required since there is a relatively small probability of an inadvertent charging pump actuation during the 8 hours before RCS depressurization is achieved and a vent established. The disabling of a charging pump is necessary since RV 203 cannot mitigate a charging/letdown mismatch event if RHR is providing decay heat removal above MODE 5 and three charging pumps are operating.

The vent must be sized ≥ 1.1 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel and to protect the RHR system from overpressurization.

The Completion Time of 8 hours to depressurize the RCS and establish a vent considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

(16)

LCO 3.0.4 only applies for entry into MODES 1, 2, 3, and 4 which includes only part of the Applicability for this LCO. Since the LTOP System helps maintain the integrity of the RCPB during low temperature conditions, it is undesirable to enter the LTOP System Applicability with no mitigation capability. This applies to both increasing or decreasing MODES. Entry into the LTOP System Applicability with both PORVs inoperable should not be made unless it is required to perform necessary repairs of the PORVs. Examples of this include a hardware related failure of both PORVs which requires breaching their integrity to restore OPERABILITY. It is undesirable to perform this type of maintenance at elevated RCS pressures with only one isolation valve available (i.e., PORV block valve). Therefore, entry into the LTOP System Applicability can be performed in order to reach a vented condition of the RCS.

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all SI pumps must be verified incapable of injecting into the RCS when the PORVs provide the RCS vent path (LCO 3.4.12.a) and a minimum of two SI pumps must be verified incapable of injecting into the RCS when the RCS is depressurized and an RCS vent ≥ 1.1 square inches is established (LCO 3.4.12.b). The SI pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the following:

- a. placing the pump control switch in the pull-stop position and closing at least one valve in the discharge flow path;

(continued)

BASES

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

- b. locking closed a manual isolation valve in the injection path; or
- c. closing a motor operated isolation valve in the injection path and removing the AC power source.

82

The flowpaths through the test connections associated with the ECCS accumulator check valves (i.e., lines containing air operated valves 839A, 839B, 840A, and 840B) and the ECCS accumulator fill lines (i.e., lines containing air operated valves 835A and 835B) do not have to be isolated for this SR since the potential mass addition from a single SI pump through these six lines is limited by the installed orifices to less than that assumed for the charging/letdown mismatch analysis.

82

The ECCS accumulator motor operated isolation valves can be verified closed by use of control board indication for valve position. This verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. If the accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the PORV setpoint.

172

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The Frequency of every 12 hours thereafter for SR 3.4.12.3 ensures that the ECCS accumulator motor operated isolation valves are maintained closed and do not result in a potential LTOP actuation.

SR 3.4.12.4

The RCS vent of ≥ 1.1 square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that cannot be locked.

(continued)

BASES

~~SURVEILLANCE
REQUIREMENTS~~

No change

~~SR 3.4.12.4 (continued)~~

- b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

REQUIREMENTS

~~SURVEILLANCE~~ ~~SR 3.4.12.4~~ (continued)

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO ~~3.4.12b~~ ~~3.4.12.b~~

(169)

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve may be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required to be removed, and the manual operator is not required to be locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a CHANNEL OPERATIONAL TEST (COT) is required every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is therefore not required.

(continued)

BASES

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.6 (continued)

A Note has been added indicating that this SR is required to be performed within 12 hours after decreasing RCS cold leg temperature to less than or equal to the LTOP enable temperature specified in the PTLR if it has not been performed within the previous 31 days. Depending on the cooldown rate, the COT may not have been performed before entry into the LTOP MODES. The test must be performed within 12 hours after entering the LTOP MODES. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

SR 3.4.12.7

Verification ~~once within 12 hours and every 31 days thereafter~~ that power is removed from each EGCS accumulator motor operated isolation valve ensures that at least two independent actions must occur before the accumulator is capable of injecting into the RCS. Since power is removed under administrative control and valve position is verified every 12 hours, the ~~performance of this surveillance once within 12 hours and every 31 day~~ frequency days thereafter will provide assurance that power is removed.

(82)

(170)

This SR is modified by a Note which states that the Surveillance is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR. If the accumulator pressure is below this limit, the LTOP limit cannot be exceeded and the surveillance is not required.

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix G.
 2. Generic Letter 88-11, "NRC Position on Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations."
 3. UFSAR, Section 5.2.2.
 4. 10 CFR 50, Section 50.46.
 5. 10 CFR 50, Appendix K.
 6. Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18," dated July 26, 1979.
 7. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors."
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BASES

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

Atomic Industry Forum (AIF) GDC 16 (Ref. 1) requires that means be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (RCPB). AIF-GDC 34 also requires that the RCPB be designed to reduce the probability of rapid propagation failures. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. The leakage detection systems support these requirements by both detecting RCS LEAKAGE and identifying the location of its source. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the plant and the public.

(continued)

BASES

BACKGROUND
(continued)

A limited amount of leakage inside containment is expected from auxiliary systems (e.g. component cooling water) that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE
SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event (Ref. 2).

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 0.5 gpm primary to secondary LEAKAGE as the initial condition. The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The assumed 0.5 gpm primary to secondary LEAKAGE is relatively inconsequential.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The SLB outside of containment is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 0.5 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident outside of containment are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). However, a lower LEAKAGE limit is assumed for all SLBs to prevent a coincident SGTR due to the large stresses placed on the SG tubes as a result of the rapid cooldown and depressurization. These stress calculations conservatively assume a tube with a 0.4 inch long through-wall crack in a location with 40% local wall thinning. The analyses demonstrate that the integrity of the selected tube is maintained with sufficient margin after the SLB. The assumed through-wall crack of 0.4 inches corresponds to 0.1 gpm leakage under normal operating conditions (Ref. 4). Therefore, the primary to secondary LEAKAGE is limited to 0.1 gpm per SG.

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO
(continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified LEAKAGE and is well within the capability of a charging pump operating at its low speed setting. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, LEAKAGE through two in-series PIVs, and primary to secondary LEAKAGE, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal return (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Each Steam Generator (SG)

Total primary to secondary LEAKAGE amounting to 0.1 gpm through each SG produces acceptable offsite doses and tube stresses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident or result in a coincident SGTR. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE. The SGs shall also be OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the in-series PIVs in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

(continued)

BASES

ACTIONS B.1
(continued)

~~With the Steam Generator Tube Surveillance Program (Specification 5.5.9) not met, integrity of steam generator tubes must be determined to be acceptable for continued operation within 4 hours and B. This Condition specifically addresses the appropriate ACTIONS to be taken in the event a non-significant Program discrepancy is discovered with the plant operating in MODES 1, 2, 3~~

~~If any RCS pressure boundary LEAKAGE exists, or if the Required Action of Condition A cannot be completed within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. Examples of this type of discrepancy include administrative (e.g., documentation of inspection results) or similar deviations which do not result in inadequate tube integrity. The 4 hour Completion Time allows a reasonable period of time for correction of administrative only problems or for the plant to contact the NRC to discuss appropriate action. The 4 hour Completion Time is based on engineering judgement.~~

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~~This Condition does not supersede the ACTIONS of Condition A in the event LEAKAGE from one or more steam generators exceeds the LCO limit. In the event this occurs, the LEAKAGE must be restored to within limits within 4 hours, or a plant shutdown commenced. This Condition is also not applicable to a situation in which integrity of the tube is questionable. In the event integrity of the tube is determined to be inadequate, this Condition is no longer applicable and Condition C of this LCO should be entered immediately since no corrective actions can be implemented during MODES 1, 2, 3 and 4.~~

C.1 and C.2

~~If any RCS pressure boundary LEAKAGE exists, or if the Required Action of Condition A or B cannot be completed within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This~~

239

(continued)

BASES

action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

(continued)

D

BASES

(continued)

BASES

ACTIONS

GB.1 and GB.2 (continued)

(231)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE which is not allowed by this LCO, would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the RCS at steady state operating conditions. Therefore, this SR is required to be performed once during the initial 12 hours of steady state operation and every 72 hours thereafter.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and volume control tank levels, makeup and letdown, and RCP seal injection and return flows.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1 (continued)

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16, Issued for comment July 10, 1967.
 2. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
 3. UFSAR, Section 15.6.3.
 4. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and Atomic Industry Forum (AIF) GDC 53 (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in-series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both in-series PIVs for a given line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through in-series valves is determined by a water inventory balance (SR 3.4.13.1) or other confirmatory tests. A known component of the identified LEAKAGE before operation begins is the least of the individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight. Prior to the required surveillance testing (SR 3.4.14.1) and water inventory balance (SR 3.4.13.1) in MODES 3 and 4; any leakage through the PIVs is considered unidentified LEAKAGE.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment (i.e., intersystem LOCA), an unanalyzed accident, that could degrade the ability for low pressure injection.

(continued)

BASES

BACKGROUND
(continued)

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core damage. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs and to identify which configurations dominate the risk profile for intersystem LOCA potential. In response to Reference 6, a plant specific evaluation of intersystem LOCAs was performed to identify the most risk significant configurations.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE
SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core damage. The dominant accident sequence in the intersystem LOCA category as identified by Reference 4 was the failure of the low pressure portion of the RHR System outside of containment. This accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent increased risk of core damage.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. In response to Reference 6, a plant specific evaluation of intersystem LOCAs was performed. PIVs in the following systems connected to the RCS were evaluated:

- a. residual heat removal (RHR);
- b. safety injection (SI); and
- c. chemical and volume control.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The evaluation of intersystem LOCAs concluded that several configurations identified in References 4 and 5 existed in the RHR and SI systems. The PIV configurations in the Chemical and Volume Control System were not identified as being risk significant due to the installed orifices in the letdown piping and the use of piping designed to RCS pressure conditions from the discharge of the positive displacement pumps to containment (Ref. 7).

The PIVs identified in the SI and RHR Systems are listed below:

- 853A RHR Inlet Check Valve to Reactor Vessel Core Deluge
- 853B RHR Inlet Check Valve to Reactor Vessel Core Deluge
- 867A SI Pump Discharge and Accumulator A Check Valve to RCS Cold Leg B
- 867B SI Pump Discharge and Accumulator B Check Valve to RCS Cold Leg A
- 877A SI Pump Discharge Check Valve to RCS Hot Leg B
- 877B SI Pump Discharge Check Valve to RCS Hot Leg A
- 878A SI Pump Discharge Isolation MOV to RCS Hot Leg B
- 878C SI Pump Discharge Isolation MOV to RCS Hot Leg A
- 878F SI Pump Discharge Check Valve to RCS Hot Leg B
- 878G SI Pump Discharge Check Valve to RCS Cold Leg B
- 878H SI Pump Discharge Check Valve to RCS Hot Leg A
- 878J SI Pump Discharge Check Valve to RCS Cold Leg A

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. This LCO only applies to those PIVs which are determined to be in the most risk significant configurations (Ref. 7) as listed in Applicable Safety Analysis. The remaining PIVs are governed by LCO 3.4.13, "RCS Operational LEAKAGE" and LCO 3.6.3, "Containment Isolation Valves ~~Boundaries~~."

(149)

(continued)

BASES

LCO
(continued)

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. A leakage rate limit based on valve size is used since this is superior to a single allowable value (Ref. 8).

Reference 9 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

In MODES 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and isolation failures are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

(continued)



BASES

ACTIONS
(continued)

A.1 and A.2

A leaking flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that isolation of the affected flow path with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation. The use of a valve other than the previously leaking PIV must include consideration that the plant may no longer be in an analyzed condition. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage due to reduced RCS pressure while reducing the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1 and SR 3.4.14.2

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve and should be based on an RCS pressure of ± 20 psig of normal system operating pressure. Leakage testing requires a stable pressure condition.

For multiple in-series PIVs, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other in-series valve meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

(69)

Testing of the check valves (877A, 877B, 878F, and 878H) and the motor operated valves (878A and 878C) identified as PIVs in the SI hot leg injection lines is to be performed at least once every 40 months. This ~~extended~~ surveillance interval is allowed since the two SI hot leg injection lines are maintained closed to address pressurized thermal shock (PTS) concerns. Each injection line is isolated by two check valves and one motor operated valve in-series which must all fail to create the potential for an intersystem LOCA. Testing of the remaining RCS PIVs in the SI and RHR systems is to be performed every 24 months, a typical refueling cycle. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 10) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1 and SR 3.4.14.2 (continued)

In addition to the periodic testing requirements, testing must be performed once after the valve has been opened by flow, exercised, or had maintenance performed on it to ensure tight reseating. This maintenance does not include minor activities such as packing adjustments which do not affect the leak tightness of the valve. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been resealed. A limit of 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. Atomic Industry Forum (AIF) GDC 53, Issued for comment July 10, 1967.
4. WASH-1400 (NUREG-75/014), "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, October 1975.
5. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," May 1980.
6. Generic Letter, "LWR Primary Coolant System Pressure Isolation Valves," dated February 23, 1980.

(continued)

BASES

REFERENCES
(continued)

7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.
 8. EG&G Report, EGG-NTAP-6175.
 9. ASME, Boiler and Pressure Vessel Code, Section XI.
 10. 10 CFR 50.55a(g).
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

Atomic Industry Forum (AIF) GDC 16 (Ref. 1) requires that means be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary (RCPB). AIF-GDC 34 (Ref. 1) also requires that the RCPB be designed to reduce the probability of rapid propagation failures. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. The leakage detection systems support these requirements by both detecting RCS LEAKAGE and identifying the location of its source.

Industry practice has shown that small water flow changes can be readily detected in contained volumes by monitoring changes in water level or in the operating frequency of a pump. The containment sump used to collect unidentified LEAKAGE (i.e., containment sump A) is monitored for level and sump pump actuation and can measure approximately a 2.0 gpm leak in one hour. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. The particulate monitor (R-11) can detect a leak of 0.013 gpm within 20 minutes assuming the presence of corrosion products. The gaseous monitor (R-12) can detect a leak of 2.0 to 10.0 gpm within 1 hour and is considered a backup to the particulate monitor. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

(continued)



BASES

BACKGROUND
(continued)

Alternative means also exist to monitor RCS LEAKAGE inside containment. These include humidity detectors, air temperature and pressure monitoring, and condensate flow rate from the air coolers. The capability of these systems to detect RCS leakage is influenced by several factors including containment free volume and detector location. These systems are most useful as alarms or indirect indicating devices available to the operators and are not required by this LCO (Ref. 2).

The leakage detection systems are also used to support identification of leakage from open systems found in containment. This includes service water and fire service water systems. Leakage from these systems is required to be monitored in response to IE Bulletin No. 80-24 (Ref. 3).

APPLICABLE
SAFETY ANALYSES

(169)

During the 1970's, the NRC began evaluating asymmetric loads that result from postulating rapid opening of double-ended ruptures of RCS piping at certain locations in PWRs. The asymmetric loads produced by the postulated breaks are the result of an assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that cause large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These asymmetric loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The resolution of USI-2 for Westinghouse PWRs was use of fracture mechanics technology for RCS piping > 10 inches diameter (Ref. 5). This technology became known as leak-before-break (LBB). Included within the LBB methodology was the requirement to have leakage detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of LBB for Ginna Station is documented in Reference 6.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leakage leak occur that is detrimental to the safety of the plant and the public. Required corrective actions are provided in LCO 3.4.13, RCS Operational LEAKAGE. The capability of the leakage detection systems was evaluated by the NRC in Reference 7.

169

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

(continued)

BASES

LCO
(continued)

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump A monitor (level or pump actuation from either sump A pump), in combination with a gaseous (R-12) or particulate (R-11) radioactivity monitor provides an acceptable minimum. Alternatively, the plant vent gaseous (R-14) or particulate (R-13) monitors may be used in place of R-12 and R-11, respectively, provided that a flowpath through normally closed valve 1590 is available and R-14A is OPERABLE.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1.1, A.1.2, and A.2

With the required containment sump A monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indications of changes in leakage. In addition to an OPERABLE gaseous or particulate atmosphere monitor, the containment air cooler condensate collection system must be verified to be OPERABLE within 24 hours, or the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. The use of the gaseous monitor (R-12) is acceptable due to the increased frequency of performing SR 3.4.13.1 or the use of the containment air cooler condensate collection system.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

(29)

The containment air cooler condensate collection system is OPERABLE if the flow paths from all four containment air coolers to their respective collection tanks are available and ~~SR 3.4.15-5a~~ CHANNEL CALIBRATION of the monitor has been performed within the last 24 months. The containment air cooler condensate collection system is provided as an option for detecting RCS leakage since SR 3.4.13.1 is not performed until after 12 hours of steady state operation. Therefore, this collection system can be used during MODE changes if the containment sump monitor is inoperable.

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Actions A.1.1, A.1.2, and A.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1.1, B.1.2, and B.2.1

With both gaseous (R-12) and particulate (R-11) containment atmosphere radioactivity monitoring instrumentation channels inoperable (and their alternatives R-13 and R-14), alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a grab sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere radioactivity monitors. The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes that at least one other form of leakage detection is available.

(continued)

D
BASES

ACTIONS

B.1.1, B.1.2, and B.2.1 (continued)

Required Actions B.1.1, B.1.2, and B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitors are inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1.1, C.1.2, C.2.1, and C.2.2

169
With the required containment sump monitor and the particulate containment atmosphere radioactivity monitor ~~(R-11)~~ ~~(R-11)~~ inoperable, the only installed means of detecting leakage is the gaseous containment atmosphere radioactivity monitor (R-12). This condition does not provide a diverse means of leakage detection. Also, the gaseous monitor can only measure between a 2.0 and 10.0 gpm leak within 1 hour which may not meet the 1.0 gpm in less than four hours detection rate required by Generic Letter 84-04 (Ref. 5).

10
The Required Actions are to analyze grab samples of the containment atmosphere or perform RCS water inventory balance, SR 3.4.13.1, at a frequency of 24 hours. The combination of the gaseous monitor and either the periodic grab samples or RCS inventory balance provide information that is adequate to detect leakage. Restoration of either of the inoperable monitors to OPERABLE status within 30 days is required to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period of time.

169
Required Actions C.1.1, C.1.2, and ~~C.2.1, and C.2.2~~ are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor and particulate containment atmosphere radioactivity monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If a Required Action of Condition A, B, or C cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

This SR requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

This SR requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

(continued)



D
BASES

SURVEILLANCE
REQUIREMENTS
(continued)

~~SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5~~ 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months considers channel reliability and operating experience has proven that this Frequency is acceptable.

89

~~SR 3.4.15.5 is modified by a Note which states that the Surveillance is only required to be performed when complying with Required Action A.2 since the containment air cooler condensate collection system is not required to be OPERABLE to meet LCO 3.4.15.~~

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16 and 34, Issued for comment July 10, 1967.
 2. Regulatory Guide 1.45.
 3. IE Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment."
 4. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.
 5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
 6. Letter from D. C. DiIanni, NRC, to R. W. Kober, RG&E, Subject: "Generic Letter 84-04," dated September 9, 1985.
 7. NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Nuclear Power Plant," December 1982.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity are provided in the SRs. DOSE EQUIVALENT I-131 is calculated using Table E-7 of Regulatory Guide 1.109 (Ref. 2). The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE
SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 0.5 gpm.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSIS
(continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the plant that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity (Ref. 4). One case assumes specific activity at $1.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 500 for a duration of four hours immediately after the accident. The second case assumes the initial reactor coolant iodine activity at $60.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/E \mu\text{Ci/gm}$ for gross specific activity.

The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. The analysis also assumes a loss of offsite power at the same time as the reactor trip following the SGTR event.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric relief valves and the main steam safety valves. This steam release continues for eight hours until the residual heat removal system is utilized for cooldown purposes. All noble gas activity in the RCS which is transported to the secondary system by the tube rupture is assumed to be immediately released to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the initial cooldown ends and the RCS system pressure stabilizes below the relief valve setpoint.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSIS
(continued)

(167)

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1 for more than ~~one week~~ ^{7 days}.

The increased permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established one week time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but they would still be within 10 CFR 100 dose guideline limits.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to 1.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to $100/E \mu\text{Ci/gm}$ (where E is the average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

(continued)

BASES (continued)

APPLICABILITY In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 8 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 8 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within one week if the limit violation resulted from normal iodine spiking.

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the DOSE EQUIVALENT I-131 is greater than the LCO limit and within the acceptable range of Figure 3.4.16-1. This allowance is provided because of the significant conservatism included in the LCO limit. Also, reducing the DOSE EQUIVALENT I-131 to within limits is accomplished through use of the Chemical and Volume Control System (CVCS) demineralizers. This cleanup operation parallels plant restart following a reactor trip which frequently results in iodine spikes due to the large step decrease in reactor power level and RCS pressure excursion. The cleanup operation can normally be accomplished within the LCO Completion Time of one week.

(continued)

BASES (continued)

ACTIONS
(continued)

B.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 specific activity is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 8 hours. The change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event. The Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If the gross specific activity is not within limit, the change within 8 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents automatically venting the SG to the environment in an SGTR event. The allowed Completion Time of 8 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1

This SR requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

(continued)

BASES (continued)

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.4.16.1 (continued)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg} \geq 500^{\circ}F$. The 7 day Frequency considers the unlikelihood of a gross fuel failure during this time.

SR 3.4.16.2

This SR is only performed in MODE 1 to ensure iodine remains within limits during normal operation and following fast power changes when fuel failure is more likely to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 10 hours after a power change $\geq 15\%$ RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for \bar{E} determination is required ~~every 184 days (6 months) with the plant operating in~~ within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 equilibrium conditions operation have elapsed since the reactor was last subcritical for at least 48 hours and every 184 days (6 months) thereafter.

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~~THE FOLLOWING TEXT WAS MOVED~~

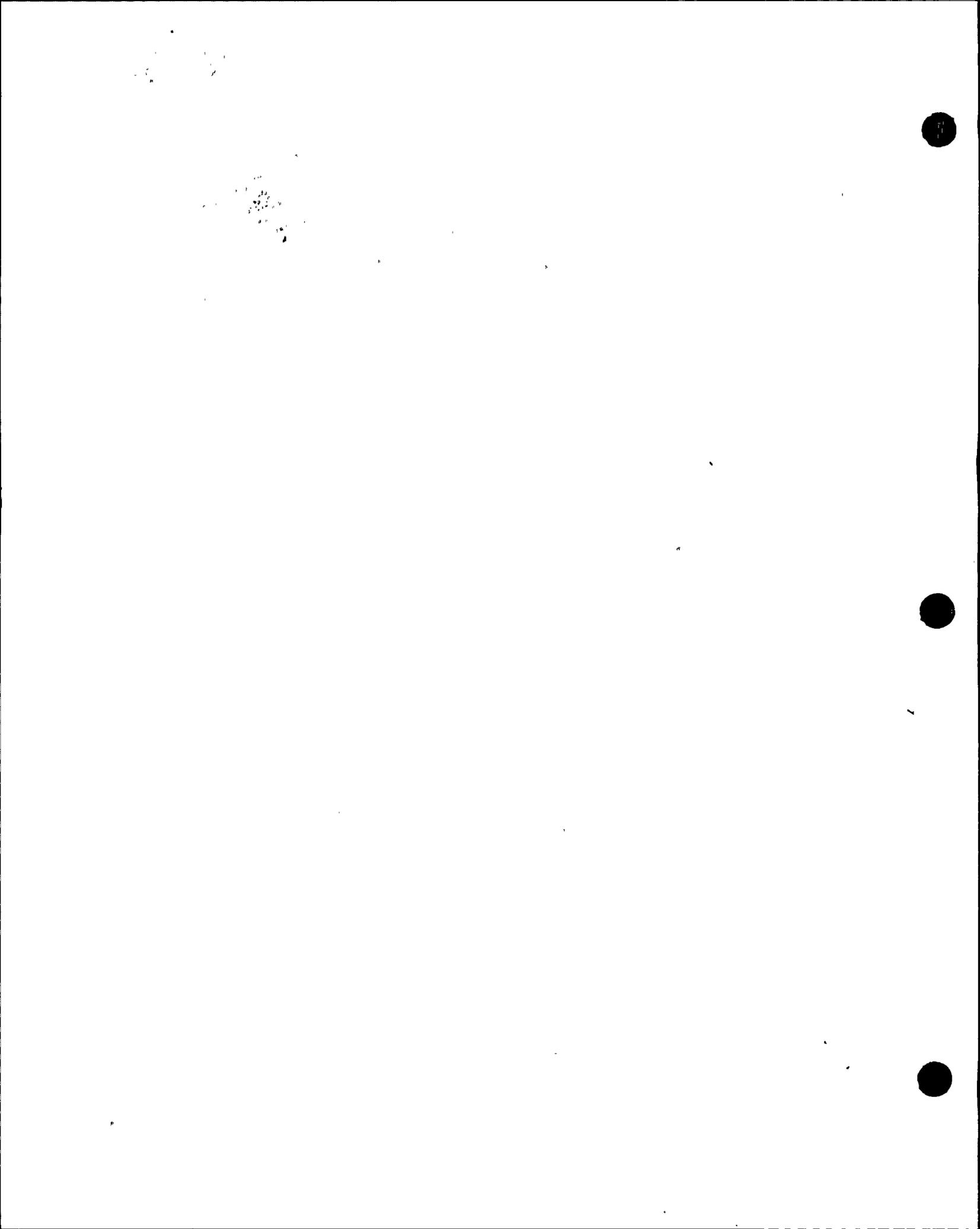
This ensures that the radioactive materials are at equilibrium so the analysis for \bar{E} is representative and not skewed by a crud burst or other similar abnormal event.

~~THE PRECEDING TEXT WAS MOVED~~

The \bar{E} determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for \bar{E} is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of ~~184 days~~ recognizes \bar{E} does not change rapidly.

This SR is modified by a Note that indicates sampling is only required to be performed in MODE 1 ~~within 31 days after~~

(continued)



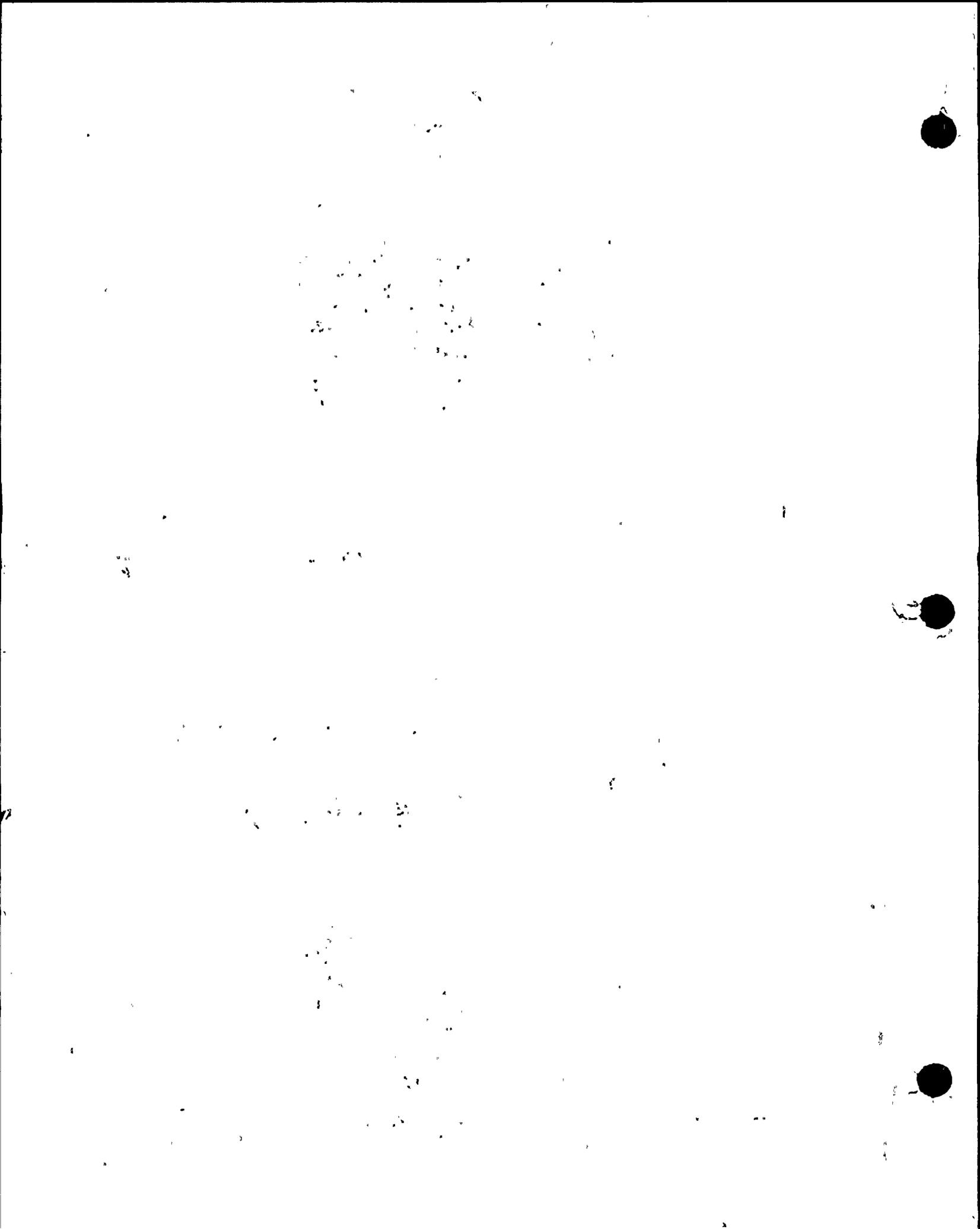
BASES (continued)

~~a minimum of 2 effective full power days and 20 days of
MODE 1 operation have elapsed since such that equilibrium
conditions are present during the reactor was last
subcritical for at least 48 hours sample.~~

174

BASES (continued)

- REFERENCES
1. 10 CFR 100.11.
 2. Regulatory Guide 1.109, Revision 1.
 3. UFSAR, Section 15.6.3.
 4. WCAP-11668, "LOFTTR2 Analysis of Potential Radiological Consequences Following a SGTR at the R.E. Ginna Nuclear Power Plant," November 1987.
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Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

Attachment L
Chapters 3.5 - 5.0
and Attachment M

Volume VI

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators:

LCO 3.5.1 Two ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1600 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce pressurizer pressure to \leq 1600 psig.	6 hours 12 hours
D. Two accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.1.1 Verify each accumulator motor operated isolation valve is fully open.	12 hours
SR 3.5.1.2 Verify borated water volume in each accumulator is ≥ 1126 cubic feet (50%) and ≤ 1154 cubic feet (82%).	12 hours
SR 3.5.1.3 Verify nitrogen cover pressure in each accumulator is ≥ 700 psig and ≤ 790 psig.	12 hours
SR 3.5.1.4 Verify boron concentration in each accumulator is within limits specified in the COLR ≥ 2100 ppm and ≤ 2600 . <i>220 ppm</i>	31 days on a STAGGERED TEST BASIS
SR 3.5.1.5 Verify power is removed from each accumulator motor operated isolation valve operator when pressurizer pressure is > 1600 psig.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - MODES 1, 2, and 3

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTES-----

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1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. Power may be restored to motor operated isolation valves ~~878A, 878B, 878C,~~ and 878D for up to 12 hours for the purpose of testing per SR 3.4.14.1 provided that power is restored to only one valve at a time.
 2. Operation in MODE 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of both RCS cold legs exceeds 375°F, whichever comes first.
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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One train inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train to OPERABLE status.	72 hours

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 4.	12 hours
C. Two trains inoperable.	C.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.2.1 Verify the following valves are in the listed position.</p> <p><u>Number</u> <u>Position</u> <u>Function</u></p> <p>825A Open RWST Suction to SI Pumps</p> <p>825B Open RWST Suction to SI Pumps</p> <p>826A Closed BAST Suction to SI Pumps</p> <p>826B Closed BAST Suction to SI Pumps</p> <p>826C Closed BAST Suction to SI Pumps</p> <p>826D Closed BAST Suction to SI Pumps</p> <p>851A Open Sump B to RHR Pumps</p> <p>851B Open Sump B to RHR Pumps</p> <p>856 Open RWST Suction to RHR Pumps</p> <p>878A Closed SI Injection to RCS Hot Leg</p> <p>878B Open SI Injection to RCS Cold Leg</p> <p>878C Closed SI Injection to RCS Hot Leg</p> <p>878D Open SI Injection to RCS Cold Leg</p> <p>896A Open RWST Suction to SI and Containment Spray</p> <p>896B Open RWST Suction to SI and Containment Spray</p>	12 hours

SURVEILLANCE	FREQUENCY
(continued)	
SR 3.5.2.2 Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3 Verify each breaker or key switch, as applicable, for each valve listed in SR 3.5.2.1, is in the correct position.	31 days
SR 3.5.2.4 Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5 Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.5.2.6 Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.5.2.7 Verify, by visual inspection, each RHR containment sump suction inlet is not restricted by debris and the containment sump screen shows no evidence of structural distress or abnormal corrosion.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - MODE 4

LCO 3.5.3 One ECCS train shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS Safety Injection (SI) subsystem inoperable.	B.1 Restore required ECCS SI subsystem to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.3.1 -----NOTE----- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ----- 3.5.2.4 The following SR is applicable for all equipment required to be OPERABLE. SR 3.5.2.4</p>	<p>In accordance with applicable SR</p>

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3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST water volume not within limits.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
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SR 3.5.4.1 Verify RWST borated water volume is \geq
300,000 gallons (88%).

~~SR 3.5.4.2~~

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SR 3.5.4.2 Verify RWST boron concentration is ≥ 2300 ppm and ≤ 2600 ppm

7 days



B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a large break loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The reactor coolant inventory is vacating the core during this phase through steam flashing and ejection out through the break. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, the core is essentially in adiabatic heatup. The balance of accumulator inventory is available to reflood the core and help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

(continued)

BASES

BACKGROUND
(continued)

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves (841 and 865) are maintained open with AC power removed under administrative control when pressurizer pressure is > 1600 psig. This feature ensures that the valves meet the single failure criterion of manually-controlled electrically operated valves per Branch Technical Position (BTP) ICSB-18 (Ref. 1). This is also discussed in References 2 and 3.

The accumulator size, water volume, and nitrogen cover pressure are selected so that one of the two accumulators is sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that one accumulator is adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure. As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for SI signal generation, the diesels starting, and the pumps being loaded and delivering full flow. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and safety injection pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the safety injection pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 5) will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty due to the reduced gas volume. A peak clad temperature penalty is an assumed increase in the calculated peak clad temperature due to a change in an input parameter. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis uses a nominal accumulator volume and includes the line water volume from the accumulator to the check valve due to these competing effects.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the time frame in which boron precipitation is addressed post LOCA. The maximum boron concentration limit is based on the coldest expected temperature of the accumulator water volume and on chemical effects resulting from operation of the ECCS and the Containment Spray (CS) System. The maximum value specified in the COLROF 2600 ppm would not create the potential for boron precipitation in the accumulator assuming a containment temperature of 60°F (Ref. 6). Analyses performed in response to 10 CFR 50.49 (Ref. 7) assumed a chemical spray solution of 2000 to 3000 ppm boron concentration (Ref. 6). The chemical spray solution impacts sump pH and the resulting effect of chloride and caustic stress corrosion on mechanical systems and components. The sump pH also affects the rate of hydrogen generation within containment due to the interaction of CS and sump fluid with aluminum components.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation at 800 psig, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 8 and 9).

The accumulators satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Two accumulators are required to ensure that 100% of the contents of one accumulator will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than one accumulator is injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 5) could be violated.

For an accumulator to be considered OPERABLE, the motor-operated isolation valve must be fully open, power removed above 1600 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1600 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

(continued)

BASES

APPLICABILITY
(continued)

This LCO is only applicable at pressures > 1600 psig. At pressures \leq 1600 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 5) limit of 2200°F.

In MODE 3, with RCS pressure \leq 1600 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood since the accumulator water volume is very small when compared to RCS and RWST inventory. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators are not expected to discharge following a large steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

(continued)

BASES

ACTIONS
(continued)

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of one accumulator cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to ≤ 1600 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both accumulators are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

(169)

SR 3.5.1.1

Each accumulator motor-operated isolation valve ^{shall} ~~should~~ be verified to be fully open every 12 hours. Use of control board indication for valve position is an acceptable verification. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

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SR 3.5.1.2 and SR 3.5.1.3

The borated water volume and nitrogen cover pressure ^{shall} ~~should~~ be verified every 12 hours for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Main control board alarms are also available for these accumulator parameters. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(169)

SR 3.5.1.4

The boron concentration ^{shall} ~~should~~ be verified to be within required limits for each accumulator every 31 days on a STAGGERED TEST Frequency since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day STAGGERED TEST Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

(continued)



BASES

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~~The boron concentration limits are specified in the COLR.~~

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is > 1600 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, no accumulators would be available for injection if the LOCA were to occur in the cold leg containing the only OPERABLE accumulator. Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.

REFERENCES

1. Branch Technical Position (BTP) ICSB-18 "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."
 2. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topics VI-7.F, VII-3, VII-6, and VIII-2," dated June 24, 1981.
 3. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
 4. UFSAR, Section 6.3.
 5. 10 CFR 50.46.
 6. UFSAR, Section 3.11.
 7. 10 CFR 50.49.
 8. UFSAR, Section 6.2.
 9. UFSAR, Section 15.6.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - MODES 1, 2, and 3

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

- a. Loss of coolant accident (LOCA) and coolant leakage greater than the capability of the normal charging system;
- b. Rod ejection accident;
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
- d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs and reactor vessel upper plenum. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sump has enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to Containment Sump B for recirculation. After approximately 20 hours, simultaneous ECCS injection is used to reduce the potential for boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

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BASES

BACKGROUND
(continued)

The ECCS flow paths which comprise the redundant trains consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the RHR pumps, heat exchangers, and the SI pumps. The RHR subsystem consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. The SI subsystem consists of three redundant, 50% capacity pumps which supply two RCS cold leg injection lines. Each injection line is capable of providing 100% of the flow required to mitigate the consequences of an accident. These interconnecting and redundant subsystem designs provide the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, suction headers supply water from the RWST to the ECCS pumps. A common supply header is used from the RWST to the safety injection (SI) and containment spray (CS) System pumps. This common supply header is provided with two in-series motor-operated isolation valves (896A and 896B) that receive power from separate sources for single failure considerations. These isolation valves are maintained open with DC control power removed via a key switch located in the control room. The removal of DC control power eliminates the most likely causes for spurious valve actuation while maintaining the capability to manually close the valves from the control room during the recirculation phase of the accident (Ref. 1). The SI pump supply header also contains two parallel motor-operated isolation valves (825A and 825B) which are maintained open by removing AC power. The removal of AC power to these isolation valves is an acceptable design against single failures that could result in undesirable component actuation (Ref. 2).

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BASES

BACKGROUND
(continued)

A separate supply header is used for the residual heat removal (RHR) pumps. This supply header is provided with a check valve (854) and motor operated isolation valve (856) which is maintained open with DC control power removed via a key switch located in the control room. The removal of DC control power eliminates the most likely causes for spurious valve actuation while maintaining the capability to manually close the valve from the control room during the recirculation phase of the accident (Ref. 3).

The three SI pumps feed two RCS cold leg injection lines. SI Pumps A and B each feeds one of the two injection lines while SI Pump C can feed both injection lines. The discharge of SI Pump C is controlled through use of two normally open parallel motor operated isolation valves (871A and 871B). These isolation valves are designed to close based on the operating status of SI Pumps A and B to ensure that SI Pump C provides the necessary flow through the RCS cold leg injection line containing the failed pump.

The discharges of the two RHR pumps and heat exchangers feed a common injection line which penetrates containment. This line then divides into two redundant core deluge flow paths each containing a normally closed motor operated isolation valve (852A and 852B) and check valve (853A and 853B) which provide injection into the reactor vessel upper plenum.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the steam generators provide core cooling until the RCS pressure decreases below the SI pump shutoff head.

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to Containment Sump B (Refs. 4 and 5). This transfer is accomplished by stopping the RHR pumps, isolating RHR from the RWST by closing motor operated isolation valve 856, opening the Containment Sump B motor operated isolation valves to RHR (850A and 850B) and then starting the RHR pumps. The SI and CS pumps are then stopped and the RWST isolated by closing motor operated isolation valve 896A and 896B for the SI and CS pump common supply header and closing motor operated isolation valve 897 or 898 for the SI pumps recirculation line.

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BASES

BACKGROUND
(continued)

The RHR pumps then supply the SI pumps if the RCS pressure remains above the RHR pump shutoff head as correlated through core exit temperature, containment pressure, and reactor vessel level indications (Ref. 6). The RHR pumps can also provide suction to the CS pumps for containment pressure control. This high-head recirculation path is provided through RHR motor operated isolation valves 857A, 857B, and 857C. These isolation valves are interlocked with valves 896A, 896B, 897, and 898. This interlock prevents opening of the RHR high-head recirculation isolation valves unless either 896A or 896B are closed and either 897 or 898 are closed. If RCS pressure is such that RHR provides adequate core and containment cooling, the SI and CS pumps remain in pull-stop. During recirculation, flow is discharged through the same paths as the injection phase. After approximately 20 hours, simultaneous injection by the SI and RHR pumps is used to prevent boron precipitation (Ref. 7). This consists of providing SI through the RCS cold legs and into the lower plenum while providing RHR through the core deluge valves into the upper plenum.

The two redundant flow paths from Containment Sump B to the RHR pumps also contain a motor operated isolation valve located within the sump (851A and 851B). These isolation valves are maintained open with power removed to improve the reliability of switchover to the recirculation phase. The operators for isolation valves 851A and 851B are also not qualified for containment post accident conditions. The removal of AC power to these isolation valves is an acceptable design against single failures that could result in an undesirable actuation (Ref. 2).

The SI subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a steam line break (SLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

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BASES

BACKGROUND
(continued)

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet AIF-GDC 44 (Ref. 8).

APPLICABLE
SAFETY ANALYSIS

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 9), will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;
- d. Core is maintained in a coolable geometry; and
- e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an SLB event and helps ensure that containment temperature limits are met post accident.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

Both ECCS subsystems are taken credit for in a large break LOCA event at full power (Refs. 6 and 10). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the pumps. The SGTR and SLB events also credit the SI pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected by the SI pumps into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core. The RHR pumps inject directly into the core barrel by upper plenum injection.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 10 and 11). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates quickly enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the SI pumps deliver sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and transferring suction to Containment Sump B. This includes securing the motor operated isolation valves as specified in SR 3.5.2.1 in position by removing the power sources as listed below.

<u>EIN</u>	<u>Position</u>	<u>Secured in Position By</u>
825A	Open	Removal of AC Power
825B	Open	Removal of AC Power
826A	Closed	Removal of AC power
826B	Closed	Removal of AC Power
826C	Closed	Removal of AC Power
826D	Closed	Removal of AC Power
851A	Open	Removal of AC power
851B	Open	Removal of AC Power
856	Open	Removal of DC Control Power
878A	Closed	Removal of AC Power
878B	Open	Removal of AC Power
878C	Closed	Removal of AC Power
878D	Open	Removal of AC Power
896A	Open	Removal of DC Control Power
896B	Open	Removal of DC Control Power

The major components of an ECCS train consists of an RHR pump and heat exchanger taking suction from the RWST (and eventually Containment Sump B), and capable of injecting through one of the two isolation valves to the reactor vessel upper plenum and one of the two lines which provide high-head recirculation to the SI and CS pumps.

(continued)

BASES

LCO
(continued)

Also included within the ECCS train are two of three SI pumps capable of taking suction from the RWST and Containment Sump B (via RHR), and injecting through one of the two RCS cold leg injection lines. In the case where SI Pump C is inoperable, both RCS cold leg injection lines must be OPERABLE to provide 100% of the ECCS flow equivalent to a single train of SI due to the location of check valves 870A and 870B.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

In MODE 4, the ECCS requirements are as described in LCO 3.5.3, "ECCS - MODE 4."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO ~~3.9.3, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft.," and LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft.," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level $<$ 23 Ft."~~

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As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room or by field test personnel. The note also allows an SI isolation MOV to

(continued)

BASES

be powered for up to 12 hours for the performance of this testing.

(continued)

BASES

APPLICABILITY
(continued)

As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," may be necessary since the LTOP arming temperature is near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 4, 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Mode 4 core cooling requirements are addressed by LCO 3.4.6, "RCS Loops - Mode 4," and LCO 3.5.3, "ECCS - Shutdown MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.3, "~~Residual Heat Removal (RHR) and Coolant Circulation High Water Level,~~" and LCO 3.9.4, "~~Residual Heat Removal (RHR) and Coolant Circulation Low Water Level~~" ~~Circulation - Water Level \geq 23 Ft.,"~~ and LCO 3.9.5, "~~Residual Heat Removal (RHR) and Coolant Circulation - Water Level $<$ 23 Ft."~~

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ACTIONS

A.1

With one train inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 12) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering 100% design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or necessary supporting systems are not available.

(continued)

BASES

ACTIONS

A.1 (continued)

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one active component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

In the case where SI Pump C is inoperable, both RCS cold leg injection lines must be OPERABLE to provide 100% of the ECCS flow equivalent to a single train of SI due to the location of check valves 870A and 870B.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 2) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

B.1 and B.2

If the inoperable train cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS
(continued)

C.1

If both trains of ECCS are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be immediately entered. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

SURVEILLANCE
REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Use of control board indication for valve position is an acceptable verification. Misalignment of these valves could render both ECCS trains inoperable. The listed valves are secured in position by removal of AC power or key locking the DC control power. These valves are operated under administrative controls such that any changes with respect to the position of the valve breakers or key locks is unlikely. The verification of the valve breakers and key locks is performed by SR 3.5.2.3. Mispositioning of these valves can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned valve is unlikely.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position in most cases, would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Verification every 31 days that AC or DC power is removed, as appropriate, for each valve specified in SR 3.5.2.1 ensures that an active failure could not result in an undetected misposition of a valve which affects both trains of ECCS. If this were to occur, no ECCS injection or recirculation would be available. Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at a single point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

REFERENCES SURVEILLANCE SR 3.5.2.7
REQUIREMENTS

(continued)

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Periodic inspections of the containment sump suction inlet to the RHR System ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.
2. Branch Technical Position (BTP) ICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically Operated Valves."
3. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 42 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant (TAC No. 79829)," dated June 3, 1991.
4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.
5. NUREG-0821.
6. UFSAR, Section 6.3.
7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic IX-4, Boron Addition System, R. E. Ginna," dated August 26, 1981.
8. Atomic Industrial Forum (AIF) GDC 44, Issued for comment July 10, 1967.

(continued)

BASES

(72) REFERENCES 9. 10 CFR 50.46.

(continued)

10. UFSAR, Section 15.6.
 11. UFSAR, Section 6.2.
 12. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - MODE 4

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - MODES 1, 2, and 3," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR).

177 The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2. The RHR subsystem must also be capable of taking suction from containment Sump B to provide recirculation.

D APPLICABLE

SAFETY ANALYSES

~~The~~ There are no Applicable Safety Analyses section of Bases 3.5.2 also which apply to the applies to this Bases section ECCS in MODE 4 due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident.

177 ~~Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA) Therefore, the ECCS operational requirements are reduced in MODE 4. It is understood in these reductions that certain automatic safety injection (SI) SI actuations are not available. In this MODE, sufficient time exists is expected for manual actuation of the required ECCS to mitigate the consequences of a DBA (Ref. - 1) This time is also required since the RHR System may be aligned to provide normal shutdown cooling while the SI System may be isolated from the RCS due to low temperature overpressure protection (LTOP) concerns.~~

~~Only~~ Therefore, only one train of ECCS is required for MODE 4. This requirement dictates that single failures are

(continued)

BASES

(177)

not considered for this LCO due to the time available for operators to respond to an accident. The ECCS trains satisfy Criterion 34 of the NRC Policy Statement.

(continued)

BASES

LCO

(177)

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA accident.

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump. The major components of an ECCS train during MODE 4 consists of an RHR pump and heat exchanger, capable of taking suction from the RWST (and eventually Containment Sump B), and able to inject through one of two isolation valves to the reactor vessel upper plenum. Also included within the ECCS train are one of three SI pumps capable of taking suction from the RWST and injecting through one of two RCS cold leg injection lines. The high-head recirculation flow path from RHR to the SI pumps is not required in the MODE 4 since there is no accident scenario which prevents depressurization to the RHR pump shutoff head prior to depletion of the RWST.

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Based on the expected time available to respond to accident conditions during MODE 4, and the configuration of the RHR and SI trains, ECCS components are OPERABLE if they are capable of being reconfigured to the injection mode (remotely or locally) within 10 minutes. This includes taking credit for an RHR pump and heat exchanger as being OPERABLE if they are being used for shutdown cooling purposes. LCO 3.4.12, "LTOP System" contains additional requirements for the configuration of the ECCS trains SI system.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

(continued)

BASES

APPLICABILITY
(continued)

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO ~~3.9.3, "Residual Heat Removal (RHR) and Coolant Circulation High Water Level," and LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation Low Water Level." and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft."~~

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ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR subsystem. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

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BASES

ACTIONS
(continued)

B.1

(177)

With no ECCS SI subsystem OPERABLE, due to the inoperability of the SI pump or flow path from the RWST, the plant is not prepared to provide high pressure response to ~~Design-Basis Events~~ ~~an accident~~ requiring SI. The 1 hour Completion Time to restore at least one SI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance description from Bases 3.5.2 apply. This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.

REFERENCES

(177)

~~None.~~ ~~WCAP 12476, "Evaluation of LOCA During Mode 3 and Mode 4 Operation for Westinghouse NSSS," November 1991.~~



B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to both trains of the ECCS and the Containment Spray (CS) System during the injection phase of a loss of coolant accident (LOCA) recovery. A common supply header is used from the RWST to the safety injection (SI) and CS pumps. A separate supply header is used for the residual heat removal (RHR) pumps. Isolation valves and check valves are used to isolate the RWST from the ECCS and CS System prior to transferring to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump based on RWST level. Use of a single RWST to supply both trains of the ECCS and CS System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

The RWST is located in the Auxiliary Building which is normally maintained between 50°F and 104°F (Ref. 1). These moderate temperatures provide adequate margin with respect to potential freezing or overheating of the borated water contained in the RWST.

During normal operation in MODES 1, 2, and 3, the safety injection (SI), RHR, and CS pumps are aligned to take suction from the RWST.

The ECCS and CS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions. The recirculation lines for the RHR and CS pumps are directed from the discharge of the pumps to the pump suction. The recirculation lines for the SI pumps are directed back to the RWST.

(continued)

BASES

BACKGROUND
(continued)

When the suction for the ECCS and CS pumps is transferred to the containment sump, the RWST and SI pump recirculation flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the Auxiliary Building and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

- a. The RWST contains sufficient borated water to support the ECCS and CS system during the injection phase;
- b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and CS pumps at the time of transfer to the recirculation mode of cooling; and
- c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in inadequate NPSH for the RHR pumps when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE
SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and CS pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 3). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of LCO 3.5.2, "ECCS - MODES 1, 2, and 3"; LCO 3.5.3, "ECCS - MODE 4"; and LCO 3.6.6, "Containment Spray (CS), Containment Recirculation Fan Cooling, and Cooler (CRFC), NaOH, and Containment Post-Accident Charcoal Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

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(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the volume required for Reactor Coolant System (RCS) makeup is a small fraction of the available RCS volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is selected such that switchover to recirculation does not occur until sufficient water has been pumped into containment to provide necessary NPSH for the RHR pumps. The minimum boron concentration is an explicit assumption in the steam line break (SLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the evaluation of chemical effects resulting from the operation of the CS System.

For a large break LOCA analysis, the minimum water volume limit of 300,000 gallons and the lower boron concentration limit are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration is used to determine the time frame in which boron precipitation is addressed post LOCA. The maximum boron concentration limit is based on the coldest expected temperature of the RWST water volume and on chemical effects resulting from operation of the ECCS and the CS System. The ~~value specified in the COLR~~ 220 ppm would not create the potential for boron precipitation in the RWST assuming an Auxiliary Building temperature of 50°F (Ref. 1). Analyses performed in response to 10 CFR 50.49 (Ref. 2) assumed a chemical spray solution of 2000 to 3000 ppm boron concentration (Ref. 1). The chemical spray solution impacts sump pH and the resulting effect of chloride and caustic stress corrosion on mechanical systems and components. The sump pH also affects the rate of hydrogen generation within containment due to the interaction of CS and sump fluid with aluminum components.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

(continued)



BASES

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and CS pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume and boron concentration limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS System OPERABILITY requirements. Since both the ECCS and the CS System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.3, "~~Residual Heat Removal (RHR) and Coolant Circulation High Water Level,~~" and LCO 3.9.4, "~~Residual Heat Removal (RHR) and Coolant Circulation Low Water Level~~ Circulation—Water Level \geq 23 Ft." and LCO 3.9.5, "~~Residual Heat Removal (RHR) and Coolant Circulation—Water Level < 23 Ft.~~"

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ACTIONS

A.1

With RWST boron concentration not within limits, it must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST boron concentration to within limits was developed considering the time required to change the boron concentration and the fact that the contents of the tank are still available for injection.

(continued)

BASES

ACTIONS
(continued)

B.1

With the RWST water volume not within limits, it must be restored to OPERABLE status within 1 hour. In this Condition, neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.5.4.1

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and CS System pump operation on recirculation. Since the RWST volume is normally stable and the RWST is located in the Auxiliary Building which provides sufficient leak detection capability, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.4.2

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

(220)

~~The RWST boron concentration limits are specified in the COLR.~~

REFERENCES

1. UFSAR, Section 3.11.
 2. 10 CFR 50.49.
 3. UFSAR, Section 6.3 and Chapter 15.
-

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Be in MODE 5.	36 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1 -----NOTE----- SR 3.0.2 is not applicable. -----</p> <p>Perform required visual examinations and leakage rate testing except for containment air lock and containment mini-purge valve testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions the Containment Leakage Rate Testing Program.</p>	<p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions the Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.2 Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.</p>	<p>In accordance with the Containment Tendon Surveillance Program</p>

35

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs on the affected air lock components.
 2. Separate Condition entry is allowed for each air lock.
 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more containment air locks with one containment air lock door inoperable.</p> <p>(34)</p>	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit through an inoperable air lock door is permissible for 7 days under administrative controls if both air locks are inoperable. <p>-----</p>	<p>(continued)</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>	
	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>	
	A.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. -----	
	Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One or more containment air locks with containment air lock interlock mechanism inoperable.</p>	<p>-----NOTES-----</p> <p>1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered.</p> <p>2. Entry and exit of containment through an air lock with an inoperable air lock interlock mechanism is permissible under the control of a dedicated individual.</p> <p>-----</p> <p>B.1 Verify an OPERABLE door is closed in the affected air lock.</p> <p><u>AND</u></p> <p>B.2 Lock an OPERABLE door closed in the affected air lock.</p> <p><u>AND</u></p> <p>B.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means.</p> <p>-----</p> <p>Verify an OPERABLE door is locked closed in the affected air lock.</p>	<p></p> <p>1 hour</p> <p>24 hours</p> <p>Once per 31 days</p>

36

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One or more containment air locks inoperable for reasons other than Condition A or B.</p> <p style="margin-left: 20px;">39</p>	<p>C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.</p> <p><u>AND</u></p> <p>C.2 Verify a door is closed in the affected air lockslock.</p> <p><u>AND</u></p> <p>C.3 Restore air lockslock to OPERABLE status.</p>	<p>Immediately</p> <p>1 hour</p> <p>24 hours</p>
<p>D. Required Action and associated Completion Time not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1</p> <p style="text-align: center;">-----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria of applicable to SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. <p style="text-align: center;">-----</p> <p>3 Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.</p> <p>SR 3.0.2 is not applicable.</p> <hr/> <p>Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p> <p>The acceptance criteria for air lock testing are:</p> <ol style="list-style-type: none"> a. Leakage rate for each air lock is $\leq 0.05 L_g$ when tested at $\geq P_g$. b. Leakage rate for each door is $\leq 0.01 L_g$ when tested at $\geq P_g$. 	<p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the Containment Leakage Rate Testing Program</p>
<p>SR 3.6.2.2</p> <p>Verify only one door in each air lock can be opened at a time.</p>	<p>24 months</p>

35



3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Barriers ~~Boundaries~~

122

LCO 3.6.3 Each containment isolation barrier ~~boundary~~ shall be OPERABLE.

-----NOTES-----

1. ~~The main steam isolation valves and~~ Not applicable to the main steam safety valves are not addressed by this LCO in MODES 1, 2, and 3.
2. Not applicable to the main steam isolation valves (MSIVs) in MODE 1, and in MODES 2 and 3 with the MSIVs open or not deactivated.
3. Not applicable to the atmospheric relief valves in MODES 1 and 2, and in MODE 3 with the Reactor Coolant System average temperature (T_{avg}) \geq 500°F.

121

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

L10

1. Penetration flow path(s), except for Shutdown Purge System valve flow paths, may be unisolated intermittently under administrative controls.
2. ~~The atmospheric relief valves are not addressed by this LCO in MODES 1 and 2, and MODE 3 with the Reactor Coolant System average temperature (T_{avg}) \geq 500°F.~~

APPLICABILITY:
MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. ~~Penetration flow path(s), except for shutdown purge flow paths, may be unisolated intermittently under administrative controls.~~
2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by failed containment isolation barriers or as a result of performing the Required Actions for this LCO boundaries.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation barrier boundary leakage results in exceeding the overall containment leakage rate acceptance criteria.
-



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------	-----------------	-----------------

A. ~~-----NOTE-----~~
Only applicable to penetration flow paths which do not use a closed system as a containment isolation boundary.

120

122

One or more penetration flow paths with one containment isolation boundary inoperable except for mini-purge valve leakage not within limit.

A. ~~One or more:~~
~~Isolate the affected penetration flow paths with path by use of at least one containment isolation barrier inoperable except for mini-purge valve leakage not within limit closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.-~~

A. ~~4 hours~~

AND

A.2 ~~-----NOTE-----~~
Isolation boundaries in high radiation areas may be verified by use of administrative means.

Once per 31 days for isolation boundaries outside containment

Verify the affected penetration flow path is isolated.

AND

Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation boundaries inside containment

1.1 Isolate the affected penetration flow path by use of at least

(continued)

4-hours

A-

122

~~Once per 31 days for
isolation devices outside
containment~~

AND

~~Prior to entering MODE 4
from MODE 5 if not
performed within the
previous 92 days for
isolation devices inside
containment~~

(continued)

ACTIONS (continued)

AB. -----2
Verify the
affected NOTE -----
Only applicable to
penetration is
isolated by an
OPERABLE flow paths
which do not use a
closed system as a
containment isolation
boundary.

120

122

One or more
penetration flow paths
with two containment
isolation boundaries
inoperable except for
mini-purge valve
leakage not within
limit.

4-hours
B.1 Isolate the affected
penetration flow path
by use of at least
one closed and
de-activated
automatic valve,
closed manual valve,
or blind flange.

(continued) B.1 h
our

One or more
penetration
flow paths
with two
containment
isolation
barriers
inoperable
except for
mini-purge
valve
leakage not
within
limit.

ACTIONS (continued)

BC. ~~-----1~~
~~Isolate the affected~~
~~affected~~ NOTE ~~-----~~
~~-----~~
 Only applicable to penetration flow path bypaths which use of at least one closed and de activated automatic valve, closed manual valve, or blind flange as a containment isolation boundary.

C.1 Isolate the affected penetration flow path by use of at least one closed and de activated automatic valve, closed manual valve, or blind flange.

1 hour

AND

C.2 ~~-----NOTE-----~~
~~-----~~
 Isolation boundaries in high radiation areas may be verified by use of administrative means.

24/2 hours

AND

B One or more penetration flow paths with one containment isolation boundary inoperable.
 Evaluate overall containment leakage rate per LCO 3.6.1.

Verify the affected penetration flow path is isolated.

AND

B.3 ~~-----~~ NOTE ~~-----~~
~~-----~~
 Isolation devices in high radiation areas may be verified by use of administrative means.

STET

Once per 31 days
~~for isolation devices outside containment~~

AND

Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

(continued)

ACTIONS (continued)

GD. One or more mini-purge penetration flow paths with one valve not within leakage limits.

120

GD.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.

24 hours

AND

D.2 ~~NOTE~~
Isolation boundaries in high radiation areas may be verified by use of administrative means.

Verify the affected penetration flow path is isolated.

Once per 31 days for isolation boundaries outside containment

AND

Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation boundaries inside containment

ACTIONS (continued)

C.2 ~~NOTE~~
~~Isolation devices in high radiation areas may be verified by use of administrative means.~~

120

~~Verify the affected penetration flow path is isolated.~~

~~Once per 31 days for isolation devices outside containment~~

AND

~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment (continued)~~

ACTIONS (continued)

~~E~~
D. (continued) One or more mini-purge penetration flow paths with two valves not within leakage limits.

120

~~D~~.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. Immediately

AND

~~D~~.2 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. 1 hour

AND

~~D.3~~ NOTE
~~Isolation devices in high radiation areas may be verified by use of administrative means.~~

~~Verify the affected penetration flow path is isolated.~~

~~Once per 31 days for isolation devices outside containment~~

AND

~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment~~

ACTIONS (continued)

F. Required Action and associated Completion Time not met.

F.1 Be in MODE 3.

AND

F.2 Be in MODE 5.

(continued) 6 hours

36 hours

120

~~E. SURVEILLANCE REQUIREMENTS~~

~~Required Action and associated Completion Time not met.~~

~~E.1 Be in MODE 3.~~

AND

~~E.2 Be in MODE 5.~~

6-hours	SURVEILLANCE REQUIREMENTS FREQUENCY
36-hours <u>SURVEILLANCE</u>	
<p>124</p> <p>SR 3.6.3.1 Verify each mini-purge valve is closed, except when the penetration flowpath(s) are permitted to be open under administrative control.</p>	SURVEILLANCES days



Be in MODE 3. _____ E.1

AND

Be in MODE 5. _____ E.2

6-hours

SURVEILLANCE
REQUIREMENTS
FREQUENCY

36-hours SURVEILLANCE

FREQUENCY SR 3.6.3.2

NOTES

1. Isolation boundaries in high radiation areas may be verified by use of administrative controls.

2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal.

Verify each containment isolation boundary that is located outside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation post accident function except for containment isolation boundaries that are open under administrative controls.

SR-
3.6.3.1

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123

SURVEILLANCE REQUIREMENTS (continued)

Be in MODE 3. _____ E.1

AND

Be in MODE 5. _____ E.2

<p>6 hours</p> <p>36 hours SURVEILLANCE</p>	<p>—</p> <p>SURVEILLANCE REQUIREMENTS FREQUENCY</p>
---	---

~~Prior to entering MODE 4 from MODE 5 if not performed within the previous 184 days~~

~~SR 3.6.3.3~~ -----NOTES-----

1. Isolation boundaries in high radiation areas may be verified by use of administrative means.

2. Not applicable to containment isolation boundaries which receive an automatic containment isolation signal.

Verify each containment isolation boundary that is located inside containment and not locked, sealed, or otherwise secured in the required position is performing its containment isolation ~~post~~ accident function, except for containment isolation boundaries that are open under administrative controls.

SR 3.6.3.3

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123

SURVEILLANCE REQUIREMENTS (continued)

~~Be in MODE 3.~~ E.1

AND

~~Be in MODE 5.~~ E.2

<p style="text-align: center;">6 hours</p> <p style="text-align: center;">36 hours SURVEILLANCE</p>	<p style="text-align: center;">— SURVEILLANCE REQUIREMENTS FREQUENCY</p>
<p>In accordance with SR 3.6.3.4 Verify the inservice Testing Program isolation time of each automatic containment isolation valve is within limits.</p>	<p>In accordance with the inservice Testing Program</p> <div style="background-color: gray; width: 100px; height: 40px; margin-top: 10px;"></div>

122

SURVEILLANCE REQUIREMENTS (continued)

~~Be in MODE 3.~~ E.1

AND

~~Be in MODE 5.~~ E.2

6 hours	-
36 hours SURVEILLANCE	SURVEILLANCE REQUIREMENTS FREQUENCY

(continued)	SR-3.6.3.4
SR 3.6.3.5 Perform required leakage rate testing of containment mini-purge valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.	p e r f e r m e d i c a t e d l e a k a g e

35

SURVEILLANCE REQUIREMENTS (continued)

~~Be in MODE 3.~~ E.1

AND

~~Be in MODE 5.~~ E.2

35

6 hours

SURVEILLANCE
REQUIREMENTS
FREQUENCY

~~36 hours~~ SURVEILLANCE

122

~~In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions SR 3.6.3.6~~

24 months

~~Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.~~

~~SR 3.6.3.5~~

~~Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.~~



3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ 1.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
127 A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	24 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LC0 3.6.5 Containment average air temperature shall be $\leq 120^{\circ}\text{F}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	24 12 hours

128

130
169

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC),
and ~~NaOH, and Containment~~ Post-Accident Charcoal Systems

169

LCO 3.6.6 Two CS trains, four CRFC units, two post-accident charcoal filter trains and the ~~spray additive tank~~ ~~NaOH system~~ shall be OPERABLE.

130

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

NOTES

42

~~Declare associated post accident charcoal filter train inoperable if CRFC unit A or C is inoperable.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CS train inoperable.	A.1 Restore CS train to OPERABLE status.	72 hours
B. One post-accident charcoal filter train inoperable.	B.1 Restore post-accident charcoal filter to OPERABLE status.	7 days
C. Two post-accident charcoal filter trains inoperable.	C.1 Restore one post-accident charcoal filter train to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>130</p> <p>D. Spray additive tank NaOH system inoperable.</p>	<p>D.1 Restore spray additive tank NaOH System to OPERABLE status.</p>	<p>72 hours</p>
(continued)		
<p>E. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p>	<p>E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.</p>	<p>6 hours 84 hours</p>
<p>42</p> <p>F. One or two CRFC units inoperable.</p>	<p>F.1 -----Restore CRFC unit(s) to OPERABLE status ---NOTE--- Required Action F.1 only required if CRFC unit A or C is inoperable. ----- Declare associated post-accident charcoal filter train inoperable. <u>AND</u> F.2 Restore CRFC unit(s) to OPERABLE status.</p>	<p>Immediately 7 days</p>
<p>G. Required Action and associated Completion Time of Condition F not met.</p>	<p>G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 5.</p>	<p>6 hours 36 hours</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)		
<p>H. Two CS trains inoperable.</p> <p><u>OR</u></p> <p>150 Spray additive tank NaOH System and one or both post-accident charcoal filter trains inoperable.</p> <p><u>OR</u></p> <p>Any combination of three three or more CRFC units inoperable.</p> <p><u>OR</u></p> <p>124 Any combination of three or more one CS and two post-accident charcoal filter trains inoperable.</p>	<p>H.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.6.1 Verify each CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position. Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.</p> <p>128</p>	<p>31 days In accordance with applicable SRs.</p>
<p>SR 3.6.6.2 Operate Verify each CRFC unit for ≥ 15 minutes. CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.</p> <p>128</p>	<p>31 days</p>
<p>SR 3.6.6.3 Operate Verify each post-accident charcoal filter train for ≥ 15 minutes. NaOH System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.</p> <p>43</p>	<p>31 days</p>
<p>SR 3.6.6.4 Verify Operate each CS pump's developed head at the flow test point is greater than or equal to the required developed head. CRFC unit for ≥ 15 minutes.</p> <p>128 43</p>	<p>31 days</p>
<p>SR 3.6.6.5 Verify cooling water flow through each CRFC unit.</p> <p>131</p>	<p>31 days</p>
<p>SR 3.6.6.6 Operate each post-accident charcoal filter train for ≥ 15 minutes.</p> <p>128 43 131</p>	<p>31 days</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.6.7 Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

(128)
 (43)
 (131)



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.6.5 — Perform required post accident charcoal filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	184 days SR 3.6.6.8
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(28)
(43)
(131)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.6.93 6.6.8 Verify each automatic CS valve in the flow path that NaOH System solution volume is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal \geq 4500 gal.</p> <p>(130) (131) (43) (128)</p>	<p>24 months 184 days</p>
<p>SR 3.6.6.103 6.6.9 Verify each CS pump starts automatically on an actual or simulated actuation signal NaOH System tank NaOH solution concentration is \geq 30% by weight.</p> <p>(131) (43) (128)</p>	<p>24 months 184 days</p>
<p>SR 3.6.6.11 Verify each CRFC unit starts automatically on an actual or simulated actuation signal 3.6.6.10 Perform required post-accident charcoal filter testing in accordance with the Ventilation Filter Testing Program (VFTP).</p> <p>(128) (131) (43)</p>	<p>24 months In accordance with the VFTP</p>
<p>SR 3.6.6.12 Verify each post accident charcoal filter train damper actuates on an actual or simulated actuation signal 3.6.6.11 Perform required CRFC unit testing in accordance with the VFTP.</p> <p>(128) (131) (43)</p>	<p>24 months In accordance with the VFTP</p>
<p>SR 3.6.6.133 6.6.12 Verify each automatic spray additive CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.</p> <p>(128) (131) (43)</p>	<p>24 -months</p>



SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.6.14 3.6.6.13 6.6.13 Verify spray additive flow rate through each eductor path CS pump starts automatically on an actual or simulated actuation signal.</p> <p>(128) (131) (43)</p>	<p>24 months</p>
<p>SR 3.6.6.14 Verify each CRFC unit starts automatically on an actual or simulated actuation signal.</p> <p>(128) (131) (43)</p>	<p>24 months</p>
<p>SR 3.6.6.15 Verify each post-accident charcoal filter train damper actuates on an actual or simulated actuation signal.</p> <p>(123) (121) (43)</p>	<p>24 months</p>
<p>(continued)</p>	
<p>SR 3.6.6.16 Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.</p> <p>(130) (44) (128) (131) (43)</p>	<p>24 months</p>
<p>SR 3.6.6.17 Verify spray additive flow through each eductor path.</p>	<p>5 years</p>
<p>SR 3.6.6.15 3.6.6.18 6.6.18 Verify each spray nozzle is unobstructed.</p> <p>(131) (43) (128)</p>	<p>10 years</p>

3.6 CONTAINMENT SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombinder inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore hydrogen recombinder to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. <u>AND</u> B.2 Restore one hydrogen recombinder to OPERABLE status.	1 hour <u>AND</u> Once per 12 hours thereafter 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Operate each hydrogen recombiner blower fan Perform a system functional check for \geq 5 minutes each hydrogen recombiner.	24 months
SR 3.6.7.2	Perform CHANNEL CALIBRATION for each hydrogen recombiner actuation and control channel.	24 months

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete containment structure, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) in accordance with Atomic Industry Forum (AIF) GDC 10 and 49 (Ref. 1). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat base mat, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Each weld seam on the inside of the liner has a leak test channel welded over it to allow independent testing of the liner when the containment is open. The liner is also insulated with closed-cell polyvinyl foam covered with metal sheeting up to the containment spray ring headers. The function of the liner insulation is to limit the mean temperature rise of the liner to only 10°F at the time associated with maximum pressure following a DBA (Ref. 2).

The containment hemispherical dome is constructed of reinforced concrete designed for all DBA related moments, axial loads, and shear forces. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. The base mat is a reinforced concrete slab that is connected to the cylinder wall by use of a hinge design which prevents the transfer of imposed shear from the cylinder wall to the base mat. This hinge consists of elastomer bearing pads located between the bottom of the cylinder wall and the base mat, and high strength steel bars which connect the cylinder walls horizontally to the base mat (Ref. 2).

(continued)



BASES (continued)

BACKGROUND
(continued)

The cylinder wall is connected to sandstone rock located beneath the containment by use of 160 post-tensioned rock anchors that are coupled with tendons located in the cylinder wall. This design ensures that the rock acts as an integral part of the containment structure.

The concrete containment structure is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the outside environment to within the limits of 10 CFR 100 (Ref. 3). SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 4), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

- a. All penetrations required to be closed during accident conditions are either:
 - 1. Capable of being closed by an OPERABLE automatic containment isolation system, or
 - 2. Closed by OPERABLE containment isolation barriersboundaries, except as provided in LCO 3.6.3, "Containment Isolation BarriersBoundaries."
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."
- c. ~~All equipment and personnel hatches or doors are closed when the air lock is not being used for entry into and exit from containment.~~

APPLICABLE
SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 5). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was originally strength tested at 69 psig (115% of design). The acceptance criteria for this test was 0.1% of the containment air weight per day at 60 psig which was based on the construction techniques that were used (Ref. 5). Following successful completion of this test, the accident analyses were performed assuming a leakage rate of 0.2% of the containment air weight per day. This leakage rate, in combination with the minimum containment engineered safeguards operating (i.e., either 2 post-accident charcoal filter trains and no containment spray, 1 post-accident charcoal filter train and 1 containment spray train, or no post-accident charcoal filter trains and 2 containment spray trains) results in offsite doses well within the limits of 10 CFR 100 (Ref. 3) in the event of a DBA.

(119)
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The leakage rate of 0.2% of the containment air weight per day is defined in 10 CFR 50, Appendix J, Option B (Ref. 5)4), as L_a : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.2% per day in the safety analysis at $P_a = 59.860$ psig (Ref. 5).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_a$ except prior to entering MODE 4 for the first time following performance of periodic testing performed in accordance with 10 CFR 50, Appendix J, Option B. At that time, the combined Type B and C leakage must be $< 0.6 L_a$ on a maximum pathway leakage rate (MXPLR) basis, and the overall Type A leakage must be $< 0.75 L_a$. At all other times prior to performing as found testing, the acceptance criteria for Type B and C testing is $< 0.6 L_a$ on a minimum pathway leakage rate (MNPLR) basis. ~~Containment in addition to leakiness considerations following a design basis LOCA, containment OPERABILITY is also defined by acceptable~~ requires structural integrity following a DBA.

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Compliance with this LCO will ensure a containment configuration, including personnel and equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and mini-purge valves with resilient seals (LCO 3.6.3) and administrative limits for individual isolation barriers/boundaries are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J for Type A, B, and C tests.

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APPLICABILITY

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In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In ~~MODES~~ ~~MODE~~ 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these ~~MODES~~ ~~this~~ ~~MODE~~. Therefore, containment is not required to be OPERABLE in ~~MODES~~ ~~MODE~~ 5 and 6 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

(continued)

BASES (continued)

ACTIONS

A.1

In the event containment is inoperable, the containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1

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Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. the Containment Leakage Rate Testing Program. 4), as modified by approved exemptions (Refs. 6 and 7).~~ Failure to meet air lock and mini-purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes these limits to be exceeded. As left leakage prior to entering MODE 4 for the first time following performance of required 10 CFR 50, Appendix J periodic testing, is required to be $< 0.6 L_a$ for combined Type B and C leakage on a MXPLR basis, and $< 0.75 L_a$ for overall Type A leakage (Ref. ~~8)5~~). At all other times between the required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_a$. This is maintained by limiting combined Type B and C leakage to $< 0.6 L_a$ on a MXPLR basis until performance of as found testing. At $\leq 1.0 L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions the Containment Leakage Rate Testing Program. Thus, SR 3.0.2 (which allows frequency extensions) does not apply.~~ These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are generally consistent with the recommendations of Regulatory Guide 1.35 (Ref. ~~9)7~~) except that tendon material tests and inspections are not required (Ref. ~~10)8~~).

(continued)

BASES (continued)

REFERENCES

1. Atomic Industry Forum, GDC 10 and 49, issued for comment July 10, 1967.
 2. UFSAR, Section 3.8.1.
 3. 10 CFR 100.
 4. 10 CFR 50, Appendix J, ~~Option B~~.
 5. UFSAR, Section 6.2.
 6. ~~Letter from DNEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0.-~~
 - 35 ~~7. Ziemann, NRC, to L. D. White, RG&E, Subject: "Amendment No. 17 to Provisional Operating License," dated March 28, 1978.~~
 - ~~7. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Completion of Appendix J Review," dated May 6, 1981.~~
 - ~~8. Regulatory Guide DG 1037.~~
 - ~~9. Regulatory Guide 1.35, Revision 2.~~
 - 169 ~~108. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Safety Evaluation Containment Vessel Tendon Surveillance Program," dated August 19, 1985.~~
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

There are two containment air locks installed at Ginna Station, an equipment hatch and a personnel hatch. Both air locks are nominally a right circular cylinder with a door at each end to allow personnel access. The two doors on each airlock are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains a double-tongue, single gasketed seal and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide a control board alarm if any door is opened. A single control board alarm exists for all four access doors. Additionally, a control board alarm is provided if high pressure exists between the two doors for either airlock.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analyses.

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES

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The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.2% of containment air weight per day (Ref. 1). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 2), as $L_a = 0.2\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_a = 59.860$ psig following a ~~DBA~~ the design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

The equipment hatch and personnel hatch containment air locks form part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate following a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

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BASES

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(continued)

118

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the 10 CFR 50, Appendix J Type B air lock leakage test (i.e., SR 3.6.2.1), and both air lock doors must be OPERABLE such that they can remain closed with leakage within acceptable limits following a DBA. The interlock allows only one door of an air lock to be opened at a time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

(continued)



BASES

~~APPLICABILITY~~ In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material. Normal entry into and exit from containment does not render the airlock inoperable.—

Nuclear

~~APPLICABILITY~~ In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. Therefore, the containment air locks are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. In MODE 5 and 6 to prevent leakage of radioactive material from containment, the probability and consequences of these events are reduced due to the pressure and temperature limitations of this MODE.—

~~ACTIONS~~ The ACTIONS Therefore, the containment air locks are modified by three Notes not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

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(continued)



BASES (continued)

ACTIONS The ACTIONS are modified by three Notes. The first Note allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

(continued)



BASES (continued)

This evaluation should be initiated immediately after declaring a containment air lock inoperable. This is required since the inoperability of an air lock may result in a significant increase in the overall containment leakage rate.

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(continued)



BASES

ACTIONS
(continued)

A.1, A.2, and A.3

169

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. If the between air lock door volume exceeds the allowed leakage criteria, and leakage is verified to be into containment (e.g., leakage through the equalizing valve), then the inner airlock door shall be declared inoperable and this Condition entered. If leakage exists ~~from containment~~ to the outside environment, then Condition C is entered. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour and may consist of verifying the control board alarm status for the airlock doors. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

(continued)



BASES

~~Required Action ACTIONS~~
~~A.1, A.2, and A.3 (continued)~~

~~Required Action A.3~~ verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

~~ACTIONS — The Required Actions have been modified by two~~
~~Notes. 1, A.2, and A.3 (continued)~~

~~The Required Actions have been modified by two Notes.~~ Note 1 specifies that Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed and Required Actions C.1, C.2, and C.3 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered to be inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note allows performing other activities (i.e., non TS-required activities) if the containment is entered, using the inoperable air lock, to perform an

(continued)

BASES

allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

(continued)

BASES

(continued)

ACTIONS B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A with the exception that both air lock doors are still OPERABLE and either door can be used to isolate the air lock penetration.

~~ACTIONS — The Required Actions have been modified by two Notes 1, B.2, and B.3 — (continued)~~

~~The Required Actions have been modified by two Notes. Note 1 specifies that Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed and Required Actions C.1, C.2, and C.3 are the appropriate remedial actions. Note 2 allows entry into and exit from containment through an air lock with an inoperable air lock interlock mechanism under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).~~

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

(continued)

BASES

ACTIONS
(continued)

C.1, C.2, and C.3

37

With one or more air locks inoperable for reasons other than those described in Condition A or B (e.g., both doors of an airlock are inoperable), Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within the limits of SR 3.6.2.1. In many instances, ~~containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the inoperable air lock door within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown (e.g., only one seal per door has failed), containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the inoperable air lock door within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown.~~ In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits due to the large margin between the airlock leakage and the containment overall leakage acceptance criteria.

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Required Action C.2 requires that one door in the affected containment air ~~locks~~ lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air ~~lock(s)~~ lock must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock and the containment overall leakage rate is acceptable.

(continued)



BASES

ACTIONS
(continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. the Containment Leakage Rate Testing Program. 2), as modified by approved exemptions (Ref. 3).~~ This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established based on industry experience. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate.

The SR has been modified frequency is as required by three Notes the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 requires that the results of this SR be evaluated against the acceptance criteria of SR 3.6.1.1 the Containment Leakage Rate Testing Program. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate.

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BASES

Note 3 states that ~~SURVEILLANCE~~ ~~SR 3.0.2 (which allows Frequency extensions) does not apply since the Frequency 3.6.2.2~~

~~REQUIREMENTS~~

~~(continued) The air lock interlock is required by Appendix J (Redesigned to prevent simultaneous opening of both doors in a single air lock.-2), as modified by approved exemptions. (Ref. 3)~~

BASES

~~SURVEILLANCE~~ ~~SR 3.6.2.2~~
~~REQUIREMENTS~~

~~(continued)~~ ~~The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock.~~ Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment airlock door is opened, this test is only required to be performed once every 24 months. The 24 month Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES

1. UFSAR, Section 6.2.1.1.
2. 10 CFR 50, Appendix J, Option B.

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3. ~~_____~~
~~_____~~
~~_____~~

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Boundaries

BASES

BACKGROUND The containment isolation boundaries form part of the containment pressure barrier and provide a means for fluid penetrations to be provided with two isolation boundaries.

Letter from DT
These isolation boundaries are either passive or active (automatic). Manual valves, check valves, de-activated automatic valves secured in their closed position, blind flanges, and closed systems are considered passive boundaries. Ziemann, NRC, Automatic valves designed to close without operator action following an accident, are considered active boundaries. Two boundaries in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses in accordance with Atomic Industry Forum (AIF) GDC 53 and 57 (Ref. White, RG&E, Subject: "Amendment No. 1"). 17 to Provisional Operating License, dated March 28, 1978. These active and passive boundaries make up the Containment Isolation System.

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(continued)

BASES

~~Containment Isolation Barriers~~
~~B 3.6.3~~

~~B 3.6 CONTAINMENT SYSTEMS~~

~~B 3.6.3 Containment Isolation Barriers~~

BASES

~~BACKGROUND~~ ~~The containment isolation barriers form part of the containment pressure barrier and provide a means for fluid penetrations to be provided with two isolation barriers. These isolation barriers or devices are either passive or active (automatic). Manual valves, check valves, de-activated automatic valves secured in their closed position, blind flanges, and closed systems are considered passive devices. Automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses in accordance with Atomic Industry Forum (AIF) GDC 53 and 57 (Ref. 1). These active and passive barriers make up the Containment Isolation System.~~

The Containment Isolation System is designed to provide isolation capability following a Design Basis Accident (DBA) for all fluid lines which penetrate containment. All major nonessential lines (i.e., fluid systems which do not perform an immediate accident mitigation function) which penetrate containment, except for the main feedwater lines, component cooling water to the reactor coolant pumps, and main steam lines, are either automatically isolated following an accident or are normally maintained closed in MODES 1, 2, 3, and 4. Automatic containment isolation valves are designed to close on a containment isolation signal which is generated by either an automatic safety injection (SI) signal or by manual actuation. The Containment Isolation System can also isolate essential lines at the discretion of the operators depending on the accident progression and mitigation. As a result, the containment isolation

(continued)



BASES

(122)

~~barriers~~ boundaries help ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a DBA.

(continued)



BASES

BACKGROUND
(continued)

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The OPERABILITY requirements for containment isolation ~~barriers~~ boundaries help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

In addition to the normal fluid systems which penetrate containment, there are two systems which can provide direct access from inside containment to the outside environment.

Shutdown Purge System (36 inch purge valves)

The Shutdown Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access below MODE 4. The supply and exhaust lines each contain one isolation valve and one double gasketed blind flange. Because of their large size, the shutdown purge valves are not qualified for automatic closure from their open position under DBA conditions. Also, due to the design of the blind flange assembly, the isolation valve is not required to be credited as a containment isolation barrier. Therefore, the blind flanges are installed in MODES 1, 2, 3, and 4 to ensure that the containment barrier is maintained (Ref. 2).

Mini-Purge System (6 inch purge valves)

The Mini-Purge System operates to:

- a. Reduce the concentration of noble gases within containment prior to and during personnel access, and
- b. Equalize internal and external pressures.

(continued)

BASES

BACKGROUND

Mini-Purge System (6 inch purge valves) (continued)

169

The system is designed with supply and exhaust lines ^{both of} which each contain two air operated isolation valves. Since the valves used in the Mini-Purge System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4; however, emphasis shall be placed on limiting purging and venting times to as low as reasonably achievable.

APPLICABLE

22

SAFETY ANALYSES

The containment isolation ~~barrier~~ boundary LCO was derived from the assumptions related to minimizing the loss of reactor inventory and establishing the containment barrier during major accidents. As part of the containment barrier, OPERABILITY of devices which act as containment isolation ~~barriers~~ boundaries supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

169

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 3). ~~In the analyses for each of these accidents, it is assumed that containment isolation barriers are either closed or function to close within the required isolation time following event initiation.~~ Other DBAs (e.g., This ensures that potential paths to the environment through containment isolation barriers (including containment mini-purge valves) are minimized. ~~g., locked rotor) result in the release of radioactive material within the reactor coolant system. In the analyses for each of these accidents, it is assumed that containment isolation boundaries are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment and other systems through containment isolation boundaries (including containment mini-purge valves) are minimized. The safety analyses assume that the Shutdown Purge System is isolated at event initiation.~~

169

(continued)

BASES

169

bounding

~~The DBA analysis assumes in the calculation of control room and offsite doses following a LOCA (rod ejection accident is assumed to be bounded), the accident analyses assume that, within 60 seconds after the accident, isolation 25% of the equilibrium iodine inventory and 100% of the containment is complete and leakage terminated except equilibrium noble gas inventory developed from maximum full power operation of the core is immediately available for the design leakage rate, leakage from containment (Ref. 1). The Mini-Purge System is assumed to be isolated within 5 seconds since these penetrations provide a direct path from containment to the outside environment (Ref. 2). The containment isolation total response time of 5 seconds or 60 seconds includes signal delay, diesel generator startup (only for motor operated valves affected by a loss of offsite power), and containment isolation valve stroke times.~~

(continued)

BASES

~~APPLICABLE~~ The Containment Isolation System is designed to provide two
~~SAFETY ANALYSES~~ in series barriers for each penetration so that no single
~~(continued)~~ credible failure or malfunction of an active component can
result in a loss of isolation or leakage that exceeds the
limits in the safety analyses. 4). The containment is
assumed to leak at the design leakage rate, L_d , for the
first 24 hours of the accident and at 50% of this leakage
rate for the remaining duration of the accident.

169

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

109.

The containment isolation boundaries ensure that the containment design leakage rate remains within L_0 by automatically isolating penetrations that do not serve post accident functions and providing isolation capability for penetrations associated with Engineered Safeguards functions. The maximum isolation time for automatic containment isolation valves is 60 seconds (Ref. 3). This isolation time is based on engineering judgement since the control room and offsite dose calculations are performed assuming that leakage from containment begins immediately following the accident with no credit for transport time or radionuclide decay. The 60 second isolation time takes into consideration the time required to drain piping of fluid which can provide an initial containment barrier before the containment isolation valves are required to close and the conservative assumptions with respect to core damage occurring immediately following the accident. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (only for motor operated valves affected by a loss of offsite power), and containment isolation valve stroke times.

A 5 second

The containment mini-purge valves are air operated valves which have isolation times shorter than 60 seconds since these penetrations may be opened and provide direct access to the outside environment. The accident analyses assume that these valves close prior to a hot rod burst (20 seconds) which occurs following a large break LOCA since the hot rod burst directly leads to higher radiation concentrations within containment. However, a shorter isolation time (5 seconds) for the mini-purge valves is used for additional conservatism (Ref. 3). The 5 second isolation total response time includes signal delay and containment isolation valve stroke times.

Containment isolation is also required for events which result in hot rod bursts but do not breach the integrity of the RCS (e.g., locked rotor accident). The isolation of containment following these events also isolates the RCS from all non-essential systems to prevent the release of radioactive material outside the RCS. The containment isolation time requirements for these events are bounded by those for the LOCA.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

12a

The Containment Isolation System is designed to provide two in series boundaries for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds the limits in the safety analyses. This system was originally designed in accordance with AIF GDC 53 (Ref. 1) which does not contain the specific design criteria specified in 10 CFR 50, Appendix A, GDC 55, 56, and 57 (Ref. 4)—5). In general, the Containment Isolation System meets the current GDC requirements; however, several penetrations differ from the GDC from the standpoint of installed valve type (e.g., check valve versus automatic isolation valve) or valve location (e.g., both containment isolation barriersboundaries are located inside containment). The evaluation of these penetrations is provided in Reference 3.

The containment isolation valvesboundaries satisfy Criterion 3 of the NRC Policy Statement.

LCO

12a

Containment isolation barriersboundaries form a part of the containment pressure barrier. The containment isolation barriersboundaries safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment barrier leakage rates during a DBA.

The barriersboundaries covered by this LCO are listed in Reference 56. These barriersboundaries consist of isolation valves (manual valves, check valves, air operated valves, and motor operated valves), pipe and end caps, closed systems, and blind flanges. There are three major categories of containment isolation barriersboundaries which are used depending on the type of penetration and the safety function of the associated piping system:

- a. Automatic containment isolation barriersboundaries which receive a containment isolation signal to close following an accident;

(continued)

BASES

LCO

(continued)

122

169

122

b. Normally closed containment isolation barriers/boundaries which are maintained closed in MODES 1, 2, 3, and 4 since they do not receive a containment isolation signal to close and the penetration is not used for normal power operation (but may be used for a long term accident mitigation function); and

c. Normally open, but nonautomatic containment isolation barriers/boundaries which are maintained open since the penetration/penetrations are required for normal power operation. Penetrations which utilize these type of isolation barriers/boundaries also contain a passive device (i.e., closed system), such that the normally open, but nonautomatic isolation boundary is only closed after the first passive boundary has failed.

~~The automatic containment isolation boundaries (i.e., valves) are considered OPERABLE when they are capable of closing within the stroke time specified in Reference 6. The normally closed containment isolation boundaries are considered OPERABLE when the manual valves are closed, air operated or motor operated valves are de-activated and secured in their closed position, check valves are closed with flow secured through the valve, blind flanges, pipe and end caps are in place, and closed systems are intact. The normally open, but nonautomatic, containment isolation boundaries (e.g., closed system), such that the normally open, but nonautomatic isolation barrier is only closed after the first passive barrier has failed.~~

~~The automatic containment isolation barriers (i.e., valves) are considered OPERABLE when they are de-activated and secured in their closed position or are capable of closing within the stroke time specified in Reference 5. The normally closed containment isolation barriers are considered OPERABLE when the manual valves are closed, air operated or motor operated valves are de-activated and secured in their closed position, check valves are closed with flow secured through the valve, blind flanges, pipe and end caps are in place, and closed systems are intact. The normally open, but nonautomatic, containment isolation barriers (e.g. check valves and manual valves) are considered OPERABLE when they are capable of being closed.~~

(continued)

BASES

In addition, both penetrations associated with the Shutdown Purge System must be isolated by a blind flange containing redundant gaskets, or a single gasketed blind flange with a de-activated automatic isolation valve (i.e., two passive barriers).

122

Containment isolation barrier boundary leakage per 10 CFR 50, Appendix J, Type B and C testing, is only addressed by LCO 3.6.1, "Containment," and is not a consideration in determination of containment isolation barrier boundary OPERABILITY.

(continued)

BASES

LCO
(continued)

This LCO provides assurance that the containment isolation barriersboundaries will perform their designed safety functions to control leakage from the containment during DBAs.

The LCO is modified by ~~two~~three Notes. The first Note states that the ~~main steam isolation valves and main steam safety valves are not addressed by this LCO~~ is not applicable to the main steam safety valves in MODES 1, 2, and 3. These valves are addressed by LCO 3.7.1, "Main Steam Safety Valves (MSSVs)," and ~~LCO 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves,"~~ which provideprovides appropriate Required ActionActions in the event these valves are declared inoperable.

The second Note states that the ~~atmospheric relief valves are not addressed by this LCO in MODES 1 and 2, and MODE 3 when the Reactor Coolant System average temperature (T_{avg}) is ≥ 500 F~~ not applicable to the main steam isolation valves (MSIVs) in MODE 1, and in MODES 2 and 3 with the MSIVs open or not deactivated. These valves are addressed by LCO 3.7.2, "Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves."

121

close

The third Note states that the atmospheric relief valves are not addressed by this LCO in MODES 1 and 2, and MODE 3 when the Reactor Coolant System average temperature (T_{avg}) is ≥ 500 F. These valves are addressed by LCO 3.7.4, "Atmospheric Relief Valves (ARVs)," which provides appropriate Required Actions in the event these valves are declared inoperable.

APPLICABILITY

201
122

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In ~~MODES~~MODE 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these ~~MODES~~this MODE. Therefore, the containment isolation barriersboundaries are not required to be OPERABLE in ~~MODES~~MODE 5 and 6.

(continued)

BASES

ACTIONS

40

The ACTIONS requirements for containment isolation boundaries during MODE 6 are modified by four Notes addressed in LCO 3.9.3, "Containment Penetrations." The first Note allows penetration flow paths, except for the shutdown-purge valve penetration flow paths, to be unisolated intermittently under administrative controls.

(continued)



BASES

ACTIONS The ACTIONS are modified by four Notes. The first Note allows penetration flow paths, except for the Shutdown Purge System valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual qualified in accordance with plant procedures at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the shutdown purge line penetration and the fact that these penetrations exhaust directly from the containment atmosphere to the outside environment, the penetration flow path containing these valves may not be opened under administrative controls.

1109

ACTIONS—A second Note has been added to provide clarification that, (continued) for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation barrier boundary. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation barrier boundaries are governed by subsequent Condition entry and application of associated Required Actions.

1102

A third Note has been added which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation barrier boundary, or as the result of performing the Required Actions described below.

(continued)



BASES

ACTIONS

(continued)

122

Finally, in the event the isolation ~~barrier~~**boundary** leakage results—
in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

(continued)



BASES

ACTIONS

~~This evaluation should be initiated immediately after declaring a containment isolation boundary inoperable. 1.1~~

(continued)

142

125

~~In the event one containment isolation barrier in one or more penetration flow paths this is inoperable (except for mini-purge valve leakage not within limit), required since the inability of an isolation boundary to close may result in a significant increase in the overall containment leakage rate if the affected penetration flow path must be isolated in series and redundant isolation boundary has a large "as-left" leakage rate associated with it.—~~

~~The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured and A.2~~

120

122

~~In the event one containment isolation boundary in one or more penetration flow paths is inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation boundary that cannot be adversely affected by a single active failure. Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1.1, the device boundary used to isolate the penetration should be the closest available one to containment. Required Action A.1.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.~~

(continued)



BASES

(continued)

BASES

ACTIONS
(continued)

(120)

(continued)

A.1.2-

~~For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Actions A.1.1, the~~

(169)

~~For affected penetration flow paths must be verified that cannot be restored to be OPERABLE status within the 4 hour Completion Time and that have been isolated on a periodic basis in accordance with Required Actions A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being isolated following a single failure will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment boundaries and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices boundaries outside containment" is appropriate considering the fact that the devices boundaries are operated under administrative controls and the probability of their misalignment is low. For the isolation devices boundaries inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices boundaries and other administrative controls that will ensure that isolation device boundary misalignment is an unlikely possibility.~~

(124)

Required Action A.1.2 is modified by a Note that applies to isolation devices boundaries located in high radiation areas and allows these devices boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices boundaries, once they have been verified to be in the proper position, is small.

(continued)



BASES

(continued)

BASES

ACTIONS

120

(continued)

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flowpaths which do not use a closed system as a containment isolation boundary. 2

~~An alternative to isolating an affected penetration with a closed and de-activated automatic valve, closed manual valve, blind flange, or a check valve with flow through the valve secured is for those penetrations which do use a closed system, Condition C provides the appropriate actions.~~

(continued)



BASES

~~For a penetration isolated in accordance with Required Action ACTIONS~~
~~B.2, OPERABILITY of the closed system can be accomplished~~
~~through normal system operation.~~

(continued)

~~With two containment leakage detection systems, surveillance systems, isolation boundaries in one or operator walkdowns more penetration flow paths inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. Closed systems The method of isolation must include the use of at least one isolation boundary that cannot be protected against pipe whip and missiles, seismic category I and safety class 2 piping adversely affected by a single active failure. Required Action Isolation boundaries that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. 2 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.~~

B.1, B.2, and B.3

124

122

~~With two containment isolation barriers in one or more penetration flow paths inoperable (except for mini-purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. Check valves and closed systems are not acceptable isolation devices in this instance since they cannot be assured to meet the design requirements of a normal containment isolation barrier. The 4 hour Completion Time is reasonable, considering the time required. Check valves and closed systems are not acceptable isolation boundaries in this instance since they cannot be assured to isolate meet the penetration and the relative importance design requirements of supporting a normal containment OPERABILITY during MODES 1, 2, 3, and 4 isolation boundary. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.~~

(continued)



BASES

(continued)

BASES

ACTIONS

~~Following completion of Required Action B.1, Bverification that the affected penetration flow path remains isolated must be performed in accordance with Required Action A.2, and B.~~

3—(continued)

~~In the event the affected penetration is isolated in accordance with Required Action Condition B is modified by a Note indicating that this Condition is only applicable to penetration flow paths which do not use a closed system as containment isolation boundary.1, the impact of using an isolation device which is not normally considered a containment isolation barrier must be evaluated with respect to the overall containment leakage rate per LCO 3.6.1. Required Action B.2 requires that acceptable Type A, B, and C leakage must be verified within 24 hours. The 24 hour Completion Time provides sufficient time to review plant records or perform necessary leakage testing on devices used to isolate the affected penetration and confirm that containment leakage remains acceptable. A Completion Time of 24 hours is appropriate considering the fact that the penetration remains isolated under administrative control, the time required to perform the leakage testing, and the margin available below 1.0 L_a as assumed in the accident analyses.~~

(124)

~~If the affected penetration is isolated in accordance with Required Action B.1 and containment remains OPERABLE per Required Action B.2, the affected penetration must be verified to be isolated on a periodic basis per Required Action B.3. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside of containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative control and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the last 92 days"~~

(continued)

BASES

(124)

~~is based on engineering judgement and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

(continued)

BASES

ACTIONS

(124)

~~B.1, B.2, and B. Required Action B~~ Condition A of this LCO addresses the condition of one containment isolation boundary inoperable in this type of penetration flow path.

For those penetrations which do not use a closed system, Condition C provides the appropriate actions.

(continued)

BASES

~~2 requires that acceptable Type A, B, and~~ **ACTIONS**

~~C leakage must be verified within 24 hours. The 24 hour Completion Time provides sufficient time to review plant records or perform necessary leakage testing on devices used to isolate the affected penetration and confirm that containment leakage remains acceptable. A Completion Time of 24 hours is appropriate considering the fact that the penetration remains isolated under administrative control, the time required?~~

(continued)

~~With one or more penetration flow paths with one containment isolation boundary inoperable, the inoperable boundary flow path must be restored to perform the leakage testing, and OPERABLE status of the margin available below 1.0 L_r as assumed in the accident analyses affected penetration flow path must be isolated.~~

224

~~If the method of isolation must include the use of at least one isolation barrier that cannot be adversely affected penetration is isolated in accordance with Required Action B by a single active failure. 1 and containment remains OPERABLE per Required Action B isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. 2, the affected penetration must be verified. A check valve may not be used to be isolated on a periodic basis per Required Action B isolate the affected penetration flow path. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. Required Action C. This Required Action does not require any testing or valve manipulation must be completed within the 72 hour Completion Time. Rather, it involves verification, through a the specified time period is reasonable considering the relative stability of the closed system walkdown (hence, that these reliability) to act as a penetration isolation devices outside of boundary and the relative importance of maintaining containment and capable of being mispositioned are in the correct position integrity during MODES 1, 2, 3, and 4. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative control and the probability of their misalignment is low in the event the affected penetration flow path is isolated in accordance with Required Action~~

This Required Action does not require any testing or device manipulation. Rather, it involves verification through a system walkdown, that these isolation boundaries capable of being mispositioned are in the correct position.

(continued)

BASES

G.I, the affected penetration flow path must be verified to be isolated on a periodic basis. For the isolation devices inside containment, the time period specified as "prior to periodic verification is necessary to entering MODE 4 from MODE 5 if not performed within the last 92 days" is based on engineering judgement and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is assure leak tightness of containment and that containment penetrations requiring isolation following an unlikely possibility accident are isolated.

(continued)



BASES

~~ACTIONS~~ ~~B~~The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the isolation boundaries are operated under administrative controls and the probability of their misalignment is low.

~~1.~~ ~~B~~Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths which use a closed system as a containment isolation boundary. ~~2.~~ ~~and~~ ~~B~~This Note is necessary since this Condition is written to specifically address those penetration flow paths which utilize a closed system as defined in Reference 7.

(continued)

BASES

ACTIONS C.1 and C.2 (continued)

124

Required Action C.3~~2~~ is modified by a Note that applies to isolation devices~~boundaries~~ located in high radiation areas and allows these devices~~boundaries~~ to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified~~verified~~). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

120

CD.1

122

In the event one or more containment mini-purge penetration flow paths contain one valve not within the mini-purge valve leakage limits, mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier~~boundary~~ that cannot be adversely affected by a single active failure. Isolation barriers~~boundaries~~ that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action CD.1 must have been demonstrated to meet the leakage requirements of SR 3-6-3.43 6-3-5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

120

126

(continued)

BASES

ACTIONS
(continued)

GD.2

120

In accordance with Required Action GD.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices ~~outside containment boundaries~~ and capable of being mispositioned are in the correct position. The Completion Time of "once every 31 days for isolation devices ~~boundaries~~ outside containment" is appropriate considering the fact that the devices ~~boundaries~~ are operated under administrative controls and the probability of their misalignment is low. For the isolation devices ~~boundaries~~ inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices ~~boundaries~~ and other administrative controls that will ensure that isolation device ~~boundary~~ misalignment is an unlikely possibility.

169

~~Required Action C.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means (e-~~

124

120

~~Required Action D.2 is modified by a Note that applies to isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices ~~boundaries~~, once they have been verified to be in the proper position, is small.~~

(continued)

BASES

ACTIONS
(continued)

120

DE.1

In the event one or more containment mini-purge penetration flow paths contain two valves not within the mini-purge valve leakage limits, Required Action DE.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current mini-purge results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both mini-purge valves have failed a leakage test or are not within the limits of SR 3-6-3-43 6-3-5. In many instances, containment remains OPERABLE per LCO 3.6.1 and it is not necessary to require restoration of the mini-purge penetration flow path within the 1 hour Completion Time specified in LCO 3.6.1 before requiring a plant shutdown. In addition, even with both valves failing the leakage test, the overall containment leakage rate can still be within limits due to the large margin between the mini-purge valve leakage and the containment overall leakage acceptance criteria.

126

DE.2 and D

Required Action E.3-

Required Action D.2 requires that the mini-purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated within 1 hour. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action DE.2 must have been demonstrated to meet the leakage requirements of SR 3-6-3-43 6-3-5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a major violation of containment does not exist.

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(continued)



BASES

ACTIONS

(126)

~~Following completion of Required Action E.2 and 1, verification that the affected penetration flow path remains isolated must be performed in accordance with Required Action D.3—(continued)~~

~~In accordance with Required Action D2.~~

(continued)

BASES

3, this penetration flow path must be verified to be isolated on a periodic basis

124

~~ACTIONS F. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur and F. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once every 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~Required Action D.3 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means (e.g., ensuring that all valve manipulations in these areas have been independently verified). Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.~~

(continued)

BASES

ACTIONS ~~E.1 and E.2~~
(continued)

120

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
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SR—3.6.3.1

~~This SR requires verification~~ensures that each nonautomatic containment isolation barrier located outside containment and not locked, sealed or otherwise secured in the required position and required to be closed immediately following an accident is closed. ~~the mini-purge valves are closed except when the valves are opened under administrative control.~~ The SR helps to ensure that post-accident leakage of radioactive fluids or gases outside of mini-purge valves are capable of closing in the containment barrier is within design limits. ~~environment following a LOCA. This SR does not require any testing or valve manipulation.~~ Therefore, these valves are allowed to be open for limited periods of time. Rather, it involves verification, through a system walkdown, ~~The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, maintenance activities, or for Surveillances that those isolation barriers outside containment and capable of being mispositioned are in~~ require the correct position valves to be open. Nonautomatic to be opened, the valves must be capable of closing under accident conditions, the containment isolation barriers include manual signal to the valves, blind flanges, pipe and end caps, and closed systems must be OPERABLE, and the effluent release must be monitored to ensure that it remains within regulatory limits. Since containment isolation barriers are maintained under administrative controls with containment isolation barrier tags installed, the probability of their misalignment is low and a 184. The

(continued)

BASES

31 day Frequency is based on the relative importance of these valves since they provide a direct path to verify their correct position is appropriate the outside environment and the administrative controls that are in place.—

(continued)

BASES

The ~~SURVEILLANCE~~ SR ~~specifies that isolation barriers that are open under administrative controls are 3.6.3.2~~

~~REQUIREMENTS~~
(continued)

This SR requires verification that each containment isolation boundary located outside containment and not locked, sealed or otherwise secured in the required to meet the SR during the time the barriers are open position is performing its containment isolation ~~post~~ accident function.

183

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.6.3.1~~ (continued)
REQUIREMENTS

The Note applies SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment isolation barriers located in high radiation areas and allows these devices to be verified closed by use of administrative means barrier is within design limits. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. This SR does not require any testing or valve manipulation. Therefore, the probability of misalignment of these isolation barriers, once they have been verified to be boundaries outside containment and capable of being mispositioned are in the proper correct position, is small.

~~SR 3.6.3.2~~

This SR requires verification that each nonautomatic containment isolation barrier located inside containment and not locked, sealed or otherwise secured in the required position and required to be normally closed immediately following an accident is closed. Includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation boundaries are maintained under administrative controls with containment isolation boundary tags installed, the probability of their misalignment is low and a 92 day Frequency to verify their correct position is appropriate. The SR specifies that isolation boundaries that are open under administrative controls are not required to meet the SR during the time the boundaries are open.

122
118

The SR is modified by two notes. The first Note applies to containment isolation boundaries located in high radiation areas and allows these boundaries to be verified closed by use of administrative means. Allowing verification by administrative means (e.g., procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation

(continued)



BASES

boundaries which receive an automatic signal since the ²
~~signal provides assurance the valve will be closed following~~
~~an accident.~~

the isolation time:
if these values are
verified by SR 3.6.34
and the boundaries
are required to be
OPERABLE.

(continued)

BASES

~~SURVEILLANCE~~ SR 3.6.3.3
~~REQUIREMENTS~~

(continued)

This SR requires verification that each containment isolation boundary located inside containment and not locked, sealed or otherwise secured in the required position and is performing its containment isolation post accident function. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment barrier is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation barriers/boundaries inside containment and capable of being mispositioned are in the correct position. ~~Nonautomatic containment isolation barriers include~~ This includes manual valves, blind flanges, pipe and end caps, and closed systems. Since containment isolation barriers/boundaries are maintained under administrative controls with containment isolation barrier/boundary tags installed, the probability of their misalignment is low and Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 18492 days" is appropriate. The SR specifies that isolation barriers/boundaries that are open under administrative controls are not required to meet the SR during the time they are open.

123

~~SURVEILLANCE~~ ~~SR 3.6.3.2~~ (continued)

REQUIREMENTS

118

103

~~The Note applies to containment isolation barriers located in high radiation areas and allows these devices to be verified closed~~ SR is modified by use of administrative means ~~two notes~~. Allowing verification by administrative means is considered acceptable, since access to these ~~The first Note applies to containment isolation boundaries located in high radiation areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons~~ allows these boundaries to be verified closed by use of administrative means. Therefore, the probability of misalignment of these isolation barriers, once they have been verified to be in their proper position, is small ~~Allowing verification by administrative means (e.~~

SR 3.6.3.3

(continued)

BASES

Verifying that the isolation time of each automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. (procedure control) is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these isolation boundaries, once they have been verified to be in the proper position, is small. The Second Note states that this SR is not applicable to containment isolation boundaries which receive an automatic signal since the signal provides assurance the valve will be closed following an accident.

(continued)

BASES

SURVEILLANCE REQUIREMENTS SR 3.6.3.4

(continued)

Verifying that the isolation time of each automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.43.6.3.5

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For containment mini-purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the outside environment), a leakage acceptance criteria of $\leq 0.05 L_a$ when tested at $\geq P_a$ is specified for each mini-purge isolation valve with resilient seals in the Containment Leakage Rate Testing Program. The Frequency of testing is also specified in 10 CFR 50, Appendix J, as modified by approved exemptions (Refs the Containment Leakage Rate Program.-

(continued)

BASES

~~6 and 7) SURVEILLANCE~~ SR 3.6.3.6
REQUIREMENTS

(continued) Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA.

BASES

~~SURVEILLANCE~~ ~~SR 3.6.3.5~~
~~REQUIREMENTS~~

~~(continued)~~

~~Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

REFERENCES

1. Atomic Industry Forum GDC 53 and 57, issued for comment July 10, 1967.
2. Branch Technical Position CSB 6-4, "Containment Purging During Normal Operation."
3. UFSAR, Section 6.2.4 and Table 6.2-15.
4. Regulatory Guide 1.4, Revision 2.
5. 10 CFR 50, Appendix A, GDC 55, 56, and 57.
56. Ginna Station Procedure A-3.3.
67. Letter from DNUREG-0800, Section 6.2.4. ~~L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Amendment No. 17 to Provisional Operating License," dated March 28, 1978.~~
7. ~~Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Completion of Appendix J Review," dated May 6, 1981.~~

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a DBA, post accident containment pressures could exceed calculated values. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment pressure outside the limits of the LCO violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses performed to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The worst case SLB generates larger mass and energy releases than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The initial pressure condition used in the containment analysis was 15.7 psia (1.0 psig). The maximum containment pressure resulting from the worst case SLB, 59.8 psig, does not exceed the containment design pressure, 60 psig.

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The containment was also designed for an ~~internal~~ external pressure load equivalent to -2.5 psig. However, internal pressure is limited to -2.0 psig based on concerns related to providing continued cooling for the reactor coolant pump motors inside containment.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2). Service Water System (LCO 3.7.8) temperature plays an important role in both maximizing and minimizing containment pressure following a DBA response.

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure. However, the lower pressure limit specified for this LCO is set at a more limiting pressure to ensure continued cooling of the reactor coolant pump motors inside containment which are required to be OPERABLE for a large portion of MODES 1, 2, 3, and 4.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 248 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 248 hour Completion Time is greater than the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. However, due to the large containment free volume and limited size of the containment Mini-Purge System, 248 hours is allowed to restore containment pressure to within limits. This is justified by the low probability of a DBA during this time period.

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B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that plant operation remains within the limits assumed in the containment analysis. This verification should normally be performed using PI-944. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

Calibration of PI-944 or other containment pressure monitoring devices should be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.
 2. 10 CFR 50, Appendix K.
-

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) and steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the Containment Spray (CS) and Containment Recirculation Fan Cooler (CRFC) Systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses to ensure that the total amount of energy within containment is within the capacity of the CS and CRFC Systems. The containment average air temperature is also an important consideration in establishing the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB which are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to the capability of the Engineered Safety Feature (ESF) systems to mitigate the accident, assuming the worst case single active failure. Consequently, the ESF systems must continue to function within the environment resulting from the DBA which includes humidity, pressure, temperature, and radiation considerations.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F. This results in a maximum containment air temperature of 374°F.

The initial temperature limit specified in this LCO is also used to establish the environmental qualification operating envelope for containment. The maximum SLB peak containment air temperature was calculated to exist for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses show that the time interval during which the containment air temperature peaked was short enough that the equipment surface temperatures remained below their design temperatures. Also, the equipment and cabling inside containment are protected against the direct effects of a SLB by concrete floors and shields. Therefore, it was concluded that the calculated transient containment air temperature following a LOCA (286°F) becomes limiting for environmental qualification reasons.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum allowable containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured and the OPERABILITY of equipment within containment is maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within the limit within 24 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 24 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. There are 6 containment air temperature indicators (TE-6031, TE-6035, TE-6036, TE-6037, TE-6038, and TE-6045) such that a minimum of three should be used for calculating the arithmetic average. The 24¹² hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24¹² hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal containment temperature condition.

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Calibration of these temperature indicators shall be performed in accordance with industry standards.

REFERENCES

1. UFSAR, Section 6.2.1.2.
 2. 10 CFR 50.49.
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B 3.6 CONTAINMENT SYSTEMS

130

B 3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC),
and ~~NaOH, and Containment~~ Post-Accident Charcoal Systems

BASES

BACKGROUND

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130

The CS and CRFC systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the CS system and the ~~System, NaOH System, and the Containment~~ Post-Accident Charcoal System connected to the CRFC units reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The CS, CRFC and ~~NaOH, and Containment~~ Post-Accident Charcoal Systems are designed to meet the requirements of Atomic Industry Forum (AIF) GDC 49, 52, 58, 59, 60, and 61 (Ref. 1). The CS, ~~NaOH, and~~ Post-Accident Charcoal Systems also are designed to limit offsite doses following a DBA within 10 CFR 100 guidelines.

The CRFC System, CS System, ~~NaOH System, and the Containment~~ Post-Accident Charcoal System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained and reduce the potential release of radioactive material, principally iodine, from the containment to the outside environment. The CS System, CRFC System, and ~~the NaOH System, and the Containment~~ Post-Accident Charcoal System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

(continued)



BASES

BACKGROUND
(continued)

130
Containment Spray System and NaOH Systems

169
The CS System consists of two redundant, 100% capacity trains. Each train includes a pump, spray headers, spray eductors, nozzles, valves, and piping (see Figure B 3.6.6-1). Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the CS System during the injection phase of operation through a common supply header shared by the safety injection (SI) system. In the recirculation mode of operation, CS pump suction can be transferred from the RWST to Containment Sump B via the residual heat removal (RHR) system.

The CS System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to scavenge fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the CS System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the residual heat removal coolers. However, the CS System can provide additional containment heat removal capability if required. Each train of the CS System provides adequate spray coverage to meet the system design requirements for containment heat removal.

The NaOH mixture is injected into the CS flowpath via a liquid eductor during the injection phase of an accident. The eductors are designed to ensure that the pH of the spray mixture is between 8.3 and 9.1. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid (Ref. 2).

(continued)

BASES

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BACKGROUND

Containment Spray System and NaOH Systems (continued)

The CS System is actuated either automatically by a containment Hi-Hi pressure signal or manually. DBAs which can generate an automatic actuation signal include the loss of coolant accident (LOCA) and steam line break (SLB). An automatic actuation opens the CS pump motor operated discharge valves (860A, 860B, 860C, and 860D), opens NaOH addition valves 836A and 836B, starts the two CS pumps, and begins the injection phase. A manual actuation of the CS System requires the operator to actuate two separate pushbuttons simultaneously on the main control board to begin the same sequence. The injection phase continues until an RWST low level alarm is received signaling the start of the recirculation phase of the accident.

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to Containment Sump B (Refs. 3 and 4). This transfer is accomplished by stopping the RHR pumps, isolating RHR from the RWST by closing motor operated valve 856, opening the Containment Sump B motor operated isolation valves to RHR (850A and 850B) and then starting the RHR pumps. The SI and CS pumps are then stopped and the RWST isolated by closing motor operated isolation valve 896A or 896B for the SI and CS pump common supply header and closing motor operated isolation valve 897 or 898 for the SI pumps recirculation line.

The RHR pumps then supply the SI pumps if the RCS pressure remains above the RHR pump shutoff head as correlated through core exit temperature, containment pressure, and reactor vessel level indications (Ref. 5). This high-head recirculation path is provided through RHR motor operated isolation valves 857A, 857B, and 857C. These isolation valves are interlocked with 896A, 896B, 897, and 898. This interlock prevents opening of the RHR high head recirculation isolation valves unless either 896A or 896B are closed and either 897 or 898 are closed. If RCS pressure is such that RHR provides adequate injection flow for core cooling, the SI pumps remain in pull-stop.

(continued)

BASES

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BACKGROUND

Containment Spray System and NaOH Systems (continued)

The CS System is only used during the recirculation phase if containment pressure increases above a pressure at which containment integrity is potentially challenged. Otherwise, the containment heat removal provided by the CRFC units and Containment Sump B (via the RHR system) is adequate to support containment heat removal needs and the limits on sump pH (Refs. 2 and 6).

Operation of the CS System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Recirculation Fan Cooler System

169

The CRFC System consists of four fan units (A, B, C, and D). Each cooling unit consists of a motor, fan, cooling coils, dampers, moisture separators, high efficiency particulate air (HEPA) filters, duct distributors and necessary instrumentation and controls (see Figure B 3.6.6-2). The moisture separators function to reduce the moisture content of the airstream to support the effectiveness of the post-accident charcoal filters. CRFC units A and D are supplied by one ESF bus while CRFC units B and C are supplied by a redundant ESF bus. All four CRFC units are supplied cooling water by the Service Water (SW) System via a common loop header. Air is drawn into the coolers through the fan and discharged into the containment atmosphere including the various compartments (e.g., steam generator and pressurizer compartments).

During normal operation, at least two fan units are typically operating. The CRFC System, operating in conjunction with other containment ventilation and air conditioning systems, is designed to limit the ambient containment air temperature during normal plant operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

(continued)



BASES

BACKGROUND

Containment Recirculation Fan Cooler System (continued)

In post accident operation following a SI actuation signal, the CRFC System fans are designed to start automatically if not already running. The discharge of CRFC units A and C then transfer to force flow through the post-accident charcoal filters. The temperature of the cooling water supplied by SW System (LCO 3.7.8) is an important factor in the heat removal capability of the fan units.

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Containment Post-Accident Charcoal System

The Containment Post-Accident Charcoal System consists of two redundant, 100% capacity trains. Each train includes an airtight plenum containing two banks of charcoal filter cells for removal of radioiodines (see Figure 3.6.6-2). Air flow enters the plenum through two holes in the bottom (one at each end), passes through the charcoal filter banks to the center, and is exhausted from the plenum through a hole in the top. Two normally closed air operated dampers isolate each post-accident charcoal filter train from CRFC units A and C (dampers 5871 and 5872 for Train A and 5874 and 5876 for Train B). A SI signal opens these dampers and closes two bypass dampers from the CRFC units (dampers 5873 for CRFC unit A and 5875 for CRFC unit C) to force flow through the post-accident charcoal filters.

APPLICABLE
SAFETY ANALYSES

169

The CS System and CRFC System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the LOCA and the SLB which are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the worst case single active failure.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 59.8 psig and the peak containment temperature is 374°F (both experienced during an SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Temperature," for a detailed discussion.) The analyses and evaluations assume a plant specific power level of 102%, one CS train and one containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 7).

The effect of an inadvertent CS actuation is not considered since there is no single failure, including the loss of offsite power, which results in a spurious CS actuation.

The modeled CS System actuation for the containment analysis is based on a response time associated with exceeding the containment Hi-Hi pressure setpoint to achieving full flow through the CS nozzles. To increase the response of the CS System, the injection lines to the spray headers are maintained filled with water. The CS System total response time of 37.5 seconds (assuming the containment Hi-Hi pressure is reached at time zero) includes diesel generator (DG) startup (for loss of offsite power), opening of the motor operated isolation valves, containment spray pump startup, and spray line filling (Ref. 8).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The modeled CRFC System actuation for the containment analysis is based upon a response time associated with exceeding the SI actuation levels to achieving full CRFC System air and safety grade cooling water flow. The CRFC System total response time of 44 seconds, includes signal delay, DG startup (for loss of offsite power), and service water pump and CRFC unit startup times (Ref. 9).

During a SLB or LOCA, a minimum of two CRFC units and one CS train are required to maintain containment peak pressure and temperature below the design limits.

169

The CS, NaOH, and Containment Post-Accident Charcoal Systems operate to reduce the release of fission product radioactivity from containment to the outside environment in the event of a DBA. The DBAs that result in a release of radioactive iodine within containment are the LOCA or a rod ejection accident (REA). In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is intact at event initiation to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine released is limited by reducing the iodine concentration present in the containment atmosphere.

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The required iodine removal capability of the CS, NaOH, and Containment Post-Accident Charcoal Systems is established by the consequences of the limiting DBA, which is a LOCA. The accident analyses (Ref. 10) assume that either two trains of CS, ~~one CS train and one post accident charcoal filter (taking suction from the NaOH System), one CS train and one post-accident charcoal filter train, or two post-accident charcoal filter trains~~ operate to remove radioactive iodine from the containment atmosphere.

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The CS System, NaOH System, CRFC System, NaOH System, and the Containment Post-Accident Charcoal System satisfy Criterion 3 of the NRC Policy Statement.

169

(continued)

208

208

208

BASES

LCO

(130)

During a DBA, a minimum of 2 CRFC units and one CS train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 8). Additionally, two CS trains ~~taking suction from the NaOH System~~, two CRFC units with post accident charcoal filters (i.e., units A and C), or one CRFC unit with post accident charcoal filters in combination with one CS train are also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two CS trains, four CRFC units, and two ~~post-accident charcoal filter trains and the spray-additive tank NaOH System~~ must be OPERABLE. Therefore, in the event of an accident, at least one CS and post-accident charcoal filter train, ~~the NaOH System~~, and two CRFC units operates, assuming the worst case single active failure occurs.

(118)

Each CS train includes a spray pump, spray headers, nozzles, valves, spray eductors, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and ~~manually~~ transferring suction to Containment Sump B via the RHR pumps.

For the ~~spray-additive tank NaOH System~~ to be OPERABLE, the volume and concentration of spray additive solution in the tank must be within limits and air operated valves 836A and 836B must be OPERABLE.

Each CRFC unit includes a motor, fan cooling coils, dampers, moisture separators, HEPA filters, duct distributors, instruments, and controls to ensure an OPERABLE flow path. For CRFC units A and C, flow through either the post-accident charcoal filter or the bypass is required for the units to be considered OPERABLE.

Each post-accident charcoal filter train includes a plenum containing charcoal filter banks and isolation dampers to ensure an OPERABLE flow path.

(continued)



BASES

~~APPLICABILITY~~ In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the CS System, CRFC System, and the Post Accident Charcoal System units A and C are also required to be OPERABLE.

(continued)



BASES

APPLICABILITY

130
In MODES 5 and 6, the probability, 2, 3, and consequences of these events are reduced due to a DBA could cause a release of radioactive material to the containment and an increase in containment pressure and temperature limitations of these MODES requiring the operation of the CS System, CRFC System, NaOH System, and the Post-Accident Charcoal System.—

Thus, the CS System, CRFC System, and the Post Accident Charcoal System are not required to be OPERABLE in MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

ACTIONS

42
The ACTIONS are modified by a Note which requires that the associated post accident charcoal filter train be declared inoperable if thus, the CS System, CRFC unit A or C is inoperable, NaOH System, and the Post-Accident Charcoal System are not required to be OPERABLE in MODES 5 and 6.—

The loss of CRFC unit

ACTIONS

A or C results in the associated post accident charcoal filter train being inoperable since the post accident charcoal filter trains do not have their own fan assembly.

A-1

With one CS train inoperable, the inoperable CS train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and CRFC units are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the CS System, the redundant iodine removal afforded by the

(continued)

12



13



14

BASES

169

Containment Post-Accident Charcoal System, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS
(continued)

B.1

With one post-accident charcoal filter train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. Each post-accident charcoal filter train is capable of providing 50% of the radioactive iodine removal requirements following a DBA. The loss of CRFC unit A or C also results in its associated post-accident charcoal filter train being inoperable since the post-accident charcoal filter trains do not have their own fan assembly. The 7 day Completion Time of Required Action B.1 to restore the inoperable post-accident charcoal filter train, including the CRFC unit, is justified considering the redundant iodine removal capabilities afforded by the CS System and NaOH Systems and the low probability of a DBA occurring during this time period.

130

C.1

With both post-accident charcoal filter trains inoperable, at least one post-accident charcoal filter train must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time to restore one inoperable post-accident charcoal filter train is justified considering the redundant iodine removal capabilities afforded by the CS System and the low probability of a DBA occurring during this time period. The inoperable post-accident charcoal filter train includes, but is not limited to inoperable CRFC units A and C.

D.1

With the ~~spray additive tank NaOH System~~ inoperable, OPERABLE status must be restored within 72 hours. The 72 hour Completion Time to restore the ~~spray additive tank NaOH System~~ is justified considering the redundant iodine removal capabilities afforded by the ~~Containment Post-Accident Charcoal System~~ and the low probability of a DBA occurring during this time period.

130

169

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

130

If the inoperable CS train, post-accident charcoal filter trains, or the ~~spray additive tank NaOH System~~ cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the inoperable component(s) and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

F.1

4a

~~With one or two CRFC units inoperable, the inoperable CRFC unit(s) must be restored to OPERABLE status within 7 days and F.2~~

169

~~With one or two CRFC units inoperable, the affected post-accident charcoal filter must be declared inoperable immediately and the inoperable CRFC unit(s) must be restored to OPERABLE status within 7 days. The inoperable components previously provided up to 100% of the containment heat removal needs and may have provided iodine removal capabilities if either CRFC unit A or C is inoperable. The 7 day Completion Time is justified considering the redundant heat removal capabilities afforded by combinations of the CS System and CRFC System and the low probability of DBA occurring during this period. If both CRFC units A and C are inoperable, then Condition C must also be entered.~~ CRFC units up to 50% of the

42

~~Required Action F.1 is modified by a Note which states that this required action is only applicable if CRFC unit A or C is inoperable. The loss of CRFC unit A or C results in the associated post-accident charcoal filter train being inoperable since the post-accident charcoal filter trains do not have their own fan assembly.~~

(continued)

BASES

(continued)

BASES

ACTIONS G.1 and G.2
(continued)

If the Required Action and associated Completion Time of Condition F of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

ACTIONS — H.1

(continued)

130
124

With two CS trains inoperable, the spray additive tank NaOH System and one or both post-accident charcoal filter trains inoperable, any combination of three or more CRFC units inoperable, or any combination of three or more CS and one CS and two post-accident charcoal filter trains inoperable, the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

178

~~Verifying the correct alignment for manual, power operated, and automatic valves in the CS flow path provides assurance that the proper flow paths will exist for CS System operation. The applicable SR descriptions from Bases 3.5.2 apply. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, is required since these were verified to be in the correct position prior to locking, sealing, or securing the OPERABILITY of valves 896A and 896B is also required for the CS System.~~

~~This SR does not require any testing or valve manipulation~~
3.6.6.2

~~Verifying the correct alignment for manual, power operated and automatic valves in the CS flow path provides assurance that the proper flow paths will exist for CS System operation. Rather, it involves verification, through a~~

(continued)

111
111



BASES

~~system walkdown. This SR does not apply to valves that these valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing.~~

~~This SR—3.6.6.2~~

~~Operating each CRFC unit for ≥ 15 minutes once every 31 days ensures that all CRFC units are OPERABLE and that all associated controls are functioning properly does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.~~

(continued)

BASES

SURVEILLANCE SR 3.6.6.3
REQUIREMENTS

(continued)

43

Verifying the correct alignment for manual, power operated, and automatic valves in the NaOH System flow path provides assurance that the proper flow paths will exist for NaOH System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.4

Operating each CRFC unit for ≥ 15 minutes once every 31 days ensures that all CRFC units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of significant degradation of the CRFC units occurring between surveillances. It has also been shown to be acceptable through operating experience.

SURVEILLANCE SR 3.6.6.3

REQUIREMENTS

(continued)

131

~~Operating each post accident charcoal filter train for ≥ 15 minutes once every 31 days ensures that all trains are OPERABLE and that all dampers are functioning properly.~~

Verifying cooling water (i.e., SW) flow to each CRFC unit provides assurance that the energy removal capability of the CRFC assumed in the accident analyses will be achieved (Ref. 11). The minimum and maximum SW flows are not required to be specifically determined by this SR due to the potential for a containment air temperature transient. Instead, this SR verifies that SW flow is available to each CRFC unit. The 31 day Frequency was developed considering the known

(continued)

BASES

reliability of the SW System, the two CRFC train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

(continued)



BASES

SURVEILLANCE SR 3.6.6.6
REQUIREMENTS

(continued)

Operating each post-accident charcoal filter train for ≥ 15 minutes once every 31 days ensures that all trains are OPERABLE and that all dampers are functioning properly. It also ensures that blockage can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the post-accident charcoal filter trains, the redundancy available, and the low probability of significant degradation of the train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.4 3.6.6.7

130

Verifying each CS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 11-12). Since the CS pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice testing confirms component OPERABILITY, trends performance, and detects incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

131

SR 3.6.6.5

~~This SR verifies that the required post-accident charcoal filter train testing is performed in accordance with the Ventilation Filter Testing Program (VFTP) 3.6.6.8~~

~~To provide effective iodine removal, the containment spray must be an alkaline solution. The VFTP includes testing charcoal absorber efficiency, minimum system flowrate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.~~

(continued)

BASES

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.6.6.6~~
~~REQUIREMENTS~~

~~(continued)~~ ~~This SR verifies that the required CRFC unit testing is performed in accordance with the VFTP. The VFTP includes testing HEPA filter performance. Specific test frequencies and additional information are discussed in detail in the VFTP.~~

~~SR 3.6.6.7~~

~~To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water that is injected. This SR is performed to verify the availability of sufficient NaOH solution in the spray additive tank. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval since the tank is normally isolated. Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.~~

(continued)

BASES

SR-3.6.6.8

(125)

~~SURVEILLANCE~~ SR-3.6.6.9

~~REQUIREMENTS~~
~~(continued)~~

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration since the tank is normally isolated and the probability that any substantial variance in tank volume will be detected.

(continued)

BASES

~~SURVEILLANCE~~ — ~~SR 3.6.6.9~~ and ~~SR 3.6.6.10~~

REQUIREMENTS

~~(continued) — These SRs require verification that each automatic CS valve~~

~~This SR verifies that the required post-accident charcoal filter train testing is performed in the flowpath (860A, 860B, 860C, and 860D) actuates to its correct position and that each CS pump starts upon receipt of an actual or simulated actuation of a containment High High pressure signal accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing charcoal adsorber efficiency, minimum system flowrate, and the physical properties of the activated charcoal. The minimum required flowrate through each of the two post-accident charcoal filters is 33,000 cubic feet per minute at accident conditions (or 38,500 cubic feet per minute at normal operating conditions). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 13).~~

(149)

(continued)

BASES

SURVEILLANCE REQUIREMENTS SR 3.6.6.11

(continued)

This SR verifies that the required CRFC unit testing is performed in accordance with the VFTP. The VFTP includes testing HEPA filter performance. The minimum required flow rate through each of the four CRFC units is 33,000 cubic feet per minute at accident conditions (or 38,500 cubic feet per minute at normal operating conditions). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 13).

150

SR 3.6.6.12 and SR 3.6.6.13

These SRs require verification that each automatic CS valve in the flowpath (860A, 860B, 860C, and 860D) actuates to its correct position and that each CS pump starts upon receipt of an actual or simulated actuation of a containment High pressure signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

132a

SR 3.6.6.113.6.6.14

This SR requires verification that each CRFC unit actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.103.6.6.12 and SR 3.6.6.113.6.6.13, above, for further discussion of the basis for the 24 month Frequency.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3-6-6-123.6.6.15

This SR requires verification every 24 months that each train of post-accident charcoal filters actuates upon receipt of an actual or simulated safety injection signal. The 24 month frequency is based on engineering judgement and has been shown to be acceptable through operating experience. See SR ~~3-6-6-93.6.6.12~~ and SR ~~3-6-6-103.6.6.13~~, above, for further discussion of the basis for the 24 month Frequency.

SR 3-6-6-133.6.6.16

13a
13b

This SR provides verification that each automatic valve in the ~~spray additive tank flowpath~~ NaOH System flow path that ~~is not locked, sealed, or otherwise secured in position~~ (836A and 836B) actuates to its correct position upon receipt of an actual or simulated actuation of a containment Hi-Hi pressure signal. The 24 month frequency is based on engineering judgement and has been shown to be acceptable through operating experience. See SR ~~3-6-6-93.6.6.12~~ and SR ~~3-6-6-103.6.6.13~~, above, for further discussion of the basis for the 24 month Frequency.

SR 3-6-6-143.6.6.17

16a

To ensure that the correct pH level is established in the borated water solution provided by the CS System, flow through the eductor is verified once every 5 years. This SR in conjunction with SR ~~3-6-6-133.6.6.16~~ provides assurance that NaOH will be added into the flow path upon CS initiation. A minimum flow of 20 gpm ~~through the eductor~~ must be established as assumed in the accident analyses. ~~A flow path must also be verified from the NaOH tank to the eductors.~~ Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow injection.

(continued)

BASES

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3-6-6-153.6.6.18

With the CS inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

13a

REFERENCES

1. Atomic Industry Forum (AIF) GDC 49, 52, 58, 59, 60, and 61, issued for comment July 10, 1967.
2. Branch Technical Position MTEB 6-1, "pH For Emergency Coolant Water For PWRs."
3. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "SEP Topic VI-7.B: ESF Automatic Switchover from Injection to Recirculation Mode, Automatic ECCS Realignment, Ginna," dated December 31, 1981.
4. NUREG-0821.
5. UFSAR, Section 6.3.
6. UFSAR, Section 6.1.2.4.
7. 10 CFR 50, Appendix K.
8. UFSAR, Section 6.2.1.2.
9. UFSAR, Section 6.2.2.2.
10. UFSAR, Section 6.5.
11. UFSAR, Section 6.2.2.1
12. ASME, Boiler and Pressure Vessel Code, Section XI.

13b

BASES

Hydrogen Recombiners
~~B-3.6.7~~

~~B-3.6 CONTAINMENT SYSTEMS~~

~~B-3.6.7 Hydrogen Recombiners~~

BASES

BACKGROUND — The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen-oxygen reaction¹³.
Regulatory Guide 1.52, Revision 2.

Figure B 3.6.6-1
Containment Spray and NaOH Systems

Figure B 3.6.6-2
CRFC and Containment Post-Accident Charcoal Systems

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by collecting the hydrogen and oxygen atmospheric mixture inside containment and oxidizing the hydrogen in a combustion chamber. Additional hydrogen is added by the recombiner to ensure that the noncondensable combustion products that could cause a progressive rise in containment pressure are avoided. Oxygen is also added to prevent depletion of oxygen below the concentration required for stable operation of the combuster. The product of combustion, water vapor, is cooled and condensed from the atmosphere by the Containment Recirculation Fan Cooler System. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA). Prevention of hydrogen accumulation during normal operation is accomplished by use of the Mini-Purge System.

(continued)

BASES

BACKGROUND
(continued)

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the Intermediate Building, a power supply from a separate Engineered Safety Features bus, and a recombiner. The recombiners are comprised of a blower fan to circulate containment air to the combuster, a combuster chamber with a main burner, two igniters (includes an installed spare), a pilot burner, and a dilution chamber downstream of the flame zone where products of the combustion are mixed with containment air to reduce the temperature of the gas leaving the system. A single recombiner is capable of maintaining the hydrogen concentration in containment at approximately 2.0 volume percent (v/o) which is below the 4.1 v/o flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence.

APPLICABLE
SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control prevents a containment wide hydrogen burn, thus ensuring the pressure and temperature inside containment as assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in containment following a LOCA as a result of:

- a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;
- b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;
- c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or
- d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 2 are used to maximize the amount of hydrogen calculated.

113
The minimum hydrogen flammability limit is 4.1 v/o, however, ~~all hydrogen must be ignited before a concentration of 6.0 v/o is reached since to avoid a dynamic overpressure overpressurization of containment could result if the hydrogen were ignited at this concentration, all hydrogen must be ignited before a concentration of 6.0 v/o is reached~~ (Ref. 3). An alternative to the ignition of hydrogen at concentrations ≥ 6.0 v/o is venting of containment using the Mini-Purge System. However, venting would most likely require evacuations of the general public within a radius of several miles surrounding the plant.

4b
Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 5.5 v/o about 31 days after the LOCA if no recombiner was functioning (Ref. 3). However, a more realistic model predicts that a hydrogen concentration of 4.1 v/o (the lower flammability limit) will be reached in 31 days. Operation of the hydrogen recombiners ~~below a concentration of 6.0 v/o will ensure ensures that containment is a concentration of 6.0 v/o would not overpressurized be reached inside containment which could result in an overpressurization given an ignition source.~~

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.1 v/o (Ref. 3).

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

Two hydrogen recombiners must be OPERABLE and capable of being placed into operation before the minimum hydrogen flammability limit of 4.1 v/o is reached following a DBA. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure. The necessary hydrogen or oxygen required to operate the hydrogen recombiner does not have to be available onsite for the hydrogen recombiner to be considered OPERABLE.

Operation with at least one hydrogen recombiner ensures that the post-LOCA hydrogen concentration can be prevented from exceeding the flammability limit or causing an overpressurization of containment given a hydrogen ignition source.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA or SLB would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a DBA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

(continued)

BASES

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the Mini-Purge System which consists of two isolation valves per penetration flow path that are capable of opening and a supply fan capable of performing purging functions. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform any Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system (e.g., opening of mini-purge valves). If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

- If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.7.1

~~Operating each hydrogen recombinder blower fan for \geq 5 minutes every 24 months ensures that the hydrogen recombiners are operational and can oxidize the hydrogen within containment following this SR requires a DBA system functional check of each hydrogen recombinder.~~

~~Operating experience has shown that these fans usually pass a functional check does not require an actual test of the Surveillance when performed at a hydrogen recombinder due to the 24 month Frequency system design which requires oxygen and hydrogen to be pumped into containment. Therefore, the Frequency was concluded. Instead, a functional check is a physical and visual inspection of the hydrogen recombiners to be acceptable from a reliability standpoint verify that piping is not plugged, the ignitor is OPERABLE, and the recombiners are not fouled. The use of a test gas (e.g., nitrogen) is acceptable. Verification that the recombiners are not fouled requires operation of the blower fan and operation of the system control valves.~~

~~The 24 month Frequency for this surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.~~

SR 3.6.7.2

This SR requires performance of a CHANNEL CALIBRATION of each hydrogen recombinder actuation and control channel. A CHANNEL CALIBRATION is required to ensure that the hydrogen recombinder will provide the correct hydrogen/oxygen mixture to the combustion chamber.

The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
 2. Safety Guide 1.7, Rev. 0.
 3. UFSAR, Section 6.2.5.
-

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Eight MSSVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MSSVs inoperable.	A.1 Restore inoperable MSSV(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY																		
<p>SR 3.7.1.1</p> <p>(170) (175)</p> <p>-----NOTE----- Only required Required to be performed prior to entry into MODE in MODES 1 and 2 from MODE 3. -----</p> <p>Verify each MSSV lift setpoint specified below in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 11\%$.</p> <p>(169)</p> <table border="1"> <thead> <tr> <th colspan="2">VALVE NUMBER</th> <th>LIFT SETTING</th> </tr> <tr> <th>SG A</th> <th>SG B</th> <th>(psig +1%, -3%)</th> </tr> </thead> <tbody> <tr> <td>3509</td> <td>3508</td> <td>1140</td> </tr> <tr> <td>3511</td> <td>3510</td> <td>1140</td> </tr> <tr> <td>3515</td> <td>3512</td> <td>1140</td> </tr> <tr> <td>3513</td> <td>3514</td> <td>1085</td> </tr> </tbody> </table>	VALVE NUMBER		LIFT SETTING	SG A	SG B	(psig +1%, -3%)	3509	3508	1140	3511	3510	1140	3515	3512	1140	3513	3514	1085	<p>In accordance with the Inservice Testing Program</p>
VALVE NUMBER		LIFT SETTING																	
SG A	SG B	(psig +1%, -3%)																	
3509	3508	1140																	
3511	3510	1140																	
3515	3512	1140																	
3513	3514	1085																	

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

LC0 3.7.2 Two MSIVs and two non-return check valves shall be OPERABLE.

APPLICABILITY: ~~MODES~~MODE 1, ~~MODE 2~~ and 3 ~~except when all MSIVs are closed and de-~~
~~activated.~~

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ACTIONS

~~NOTE~~

~~Separate Condition entry is allowed for each valve.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV or more valves inoperable in flowpath from a steam generator (SG). in MODE 1	A.1 Close inoperable MSIV Restore valve(s) to OPERABLE status. AND A-	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>24 hours B.1 Isolate affected main steam line. B. One non return check valve inoperable. 24 hours Once per 31 days 134 Verify MSIV is closed. C. Required Action and associated Completion Time of Condition B or C not met One or more valves inoperable in flowpath from a SG in MODE 2 or 3.</p>	<p>C.1 Be in MODE 3 Close MSIV. AND C.2 Be in MODE 4 Verify MSIV is closed.</p>	<p>6 hours 12 hours 8 hours Once per 7 days</p>

—(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more valves inoperable in flowpath from each steam generator Required Action and Associated Completion Time of Condition C not met.	D.1 Enter LCO 3.0.3 Be in MODE 3. AND D.2 Be in MODE 4	Immediately 6 hours 12 hours
E. One or more valves inoperable in flowpath from each SG.	E.1 Enter LCO 3.0.3	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 Verify closure time of each MSIV is ≤ 5 seconds under no flow and no load conditions.	In accordance with the Inservice Testing Program
SR 3.7.2.2 Verify each main steam non-return check valve can close.	In accordance with the Inservice Testing Program
SR 3.7.2.3 Verify each MSIV can close on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.3 ~~Main Feedwater Pump Discharge Valves (MFPDVs), Main Feedwater
Regulating Valves (MFRVs), and Associated Bypass Valves
and Main Feedwater Pump Discharge Valves (MFPDVs)~~

LCO 3.7.3 Two ~~MFPDVs, two MFRVs and~~ associated bypass valves ~~and two
MFPDVs~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 ~~except when both steam generators are
isolated from both main feedwater pumps.~~

ACTIONS

-NOTE-

Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFPDV(s) inoperable.	A.1 Close inoperable MFPDV(s).	24 hours
	<u>AND</u> A.2 Verify MFPDV(s) is closed.	Once per 317 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. B One or more MFRV(s) inoperable.</p>	<p>B.1 Close inoperable MFRV(s).</p> <p><u>AND</u></p> <p>B.2 Verify MFRV(s) is closed.</p> <p>B.1 Close or isolate MFRV(s).</p> <p><u>AND</u></p> <p>B.2 Verify MFRV(s) is closed or isolated.</p>	<p>24 hours</p> <p>Once per 31 days</p>
(continued)		
<p>135 C. One or more MFRV bypass valve(s) inoperable.</p>	<p>C.1 Close or isolate MFRV bypass valve(s).</p> <p><u>AND</u></p> <p>C.2 Verify MFRV bypass valve(s) is closed or isolated.</p>	<p>24 hours</p> <p>Once per 31 days</p>
<p>D. Required Action and associated Completion Time for Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>



MFRVs, Associated Bypass Valves, and MFPDVs
3.7.3

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. One or more MFPDV(s) and one or more MFRV(s) inoperable.</p> <p><u>OR</u></p> <p>164 One or ^{more} MFPDV(s) and one or more MFRV bypass valve(s) inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.3.1 Verify the closure time of each MFPDV is \leq 80 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2 Verify the closure time of each MFRV and associated bypass valve is \leq 10 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

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3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Two ARVs ~~ARV lines~~ shall be OPERABLE.

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APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System average temperature (T_{avg})
 $\geq 500^{\circ}F$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARV line inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore ARV line to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3 with $T_{avg} < 500^{\circ}F$.	8 hours
C. Two ARVs inoperable. C. Two ARV lines inoperable.	C.1 Enter LCO 3.0.3.	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Perform a complete cycle of each ARV.	24 months
<p>3.7 PLANT SYSTEMS</p> <p>3.7.5 Auxiliary Feedwater (AFW) System</p> <p>LCO 3.7.5 Three AFW trains and two standby AFW (SAFW) trains shall be OPERABLE SR 3.7.4.2 Verify one complete cycle of each ARV block valve.</p> <p>APPLICABILITY: MODES 1, 2, and 3.</p> <p>ACTIONS</p>	<p>24 months</p>

CONDITIONS 3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

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 IGO 3.7.5 Two motor driven AFW (MDAFW) trains, one turbine driven AFW (TDAFW) train, and two standby AFW (SAFW) trains shall be OPERABLE

APPLICABILITY: MODES 1, 2, and 3

ACTIONS
 REQUIRED ACTION

	A-REQUIRED ACTION One turbine driven AFW train flowpath inoperable.	A-COMPLETION TIME Restore turbine driven AFW flowpath or motor driven AFW train to OPERABLE status.
COMPLETION TIME CONDITION	OR One motor driven AFW train inoperable.	
A: One TDAFW train flowpath inoperable.	A.1: Restore TDAFW train flowpath to OPERABLE status.	7 days

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REQUIRED ACTION

<p>COMPLETION TIME CONDITION</p>	<p>A. REQUIRED ACTION One turbine driven AFW train flowpath inoperable.</p> <p><u>OR</u></p> <p>One motor driven AFW train inoperable.</p>	<p>A. COMPLETION TIME Restore turbine driven AFW flowpath or motor driven AFW train to OPERABLE status.</p>
<p>B. Turbine driven AFW MDAFW train inoperable.</p> <p><u>OR</u></p> <p>Two motor driven AFW trains inoperable.</p> <p><u>OR</u></p> <p>One turbine driven AFW train flowpath and one motor driven AFW train inoperable to opposite steam generators (SGs).</p>	<p>B.1 Restore MDAFW train to OPERABLE status.</p>	<p>B. 7 days Restore one AFW train or turbine driven AFW train flowpath to OPERABLE status.</p>
<p>C. TDAFW train inoperable.</p> <p><u>OR</u></p> <p>Two MDAFW trains inoperable.</p> <p><u>OR</u></p> <p>One TDAFW train flowpath and one MDAFW train inoperable to opposite steam generators (SGs).</p>	<p>C.1 Restore one MDAFW train or TDAFW train flowpath to OPERABLE status.</p>	<p>72 hours</p>



ACTIONS (continued)

REQUIRED ACTION

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COMPLETION TIME/CONDITION	A. REQUIRED ACTION One turbine driven AFW train flowpath inoperable. OR One motor driven AFW train inoperable.	A. COMPLETION TIME Restore turbine driven AFW flowpath or motor driven AFW train to OPERABLE status.

(continued)

<p>CD. All AFW trains to one or more SGs inoperable.</p>	<p>D.1 Restore one AFW train or TDAFW Flowpath to each affected SG to OPERABLE status.</p>	<p>4 hours</p>
<p>DE. One SAFW train inoperable.</p>	<p>DE.1 Restore SAFW train to OPERABLE status.</p>	<p>14 days</p>
<p>DF. Both SAFW trains inoperable.</p>	<p>DF.1 Restore one SAFW train to OPERABLE status.</p>	<p>7 days</p>

ACTIONS (continued)

REQUIRED ACTION

COMPLETION TIME CONDITION	<p>A. REQUIRED ACTION One turbine driven AFW train flowpath inoperable.</p> <p style="text-align: center;"><u>OR</u></p> <p>One motor driven AFW train inoperable.</p>	<p>A. COMPLETION TIME Restore turbine driven AFW flowpath or motor driven AFW train to OPERABLE status.</p>
<p>EG. All AFW trains Required Action and flowpaths to one associated Completion Time for Condition A, B, C, D, E, or more SGs inoperable not met.</p>	<p>EG.1 Restore one AFW train or flowpath to each affected SG to OPERABLE status. Be in MODE 3.</p> <p><u>AND</u></p> <p>G.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>F. Required Action and associated Completion Time for Condition A, B, C, D, or E not met.</p> <p>4 hours² Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>

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ACTIONS (continued)

REQUIRED ACTION

COMPLETION TIME CONDITION	A. REQUIRED ACTION One turbine driven AFW train flowpath inoperable. <u>OR</u> One motor driven AFW train inoperable.	A. COMPLETION TIME Restore turbine driven AFW flowpath or motor driven AFW train to OPERABLE status.
<p>GH. Three AFW trains and both SAFW trains inoperable.</p> <p>139</p> <p>169</p> <p>M.DAFW, T.DAFW,</p>	<p>GH.1 -----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW or SAFW train is restored to OPERABLE status. ----- Initiate action to restore one AFW or SAFW train to OPERABLE status.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.1 Verify each AFW and SAFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.5.2 -----NOTE----- Only required Required to be performed met prior to entering MODE 1 for the turbine driven AFW/DAFW pump. -----</p> <p>Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.3 Verify the developed head of each SAFW pump at the flow test point is greater than or equal to the required developed head.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.4 Perform a complete cycle of each AFW and SAFW motor operated suction valve from the Service Water System, each AFW and SAFW discharge motor operated isolation valve, and each SAFW cross-tie motor operated valve.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.5.5 Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	<p>24 months</p>

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
(continued)	
<p>SR 3.7.5.6</p> <p>-----NOTE----- Only required Required to be performed met prior to entering MODE 1 for the turbine driven AFWDAFW pump.</p> <p>-----</p> <p>Verify each AFW pump starts automatically on an actual or simulated actuation signal.</p>	<p>24 months</p>
<p>SR 3.7.5.7</p> <p>Verify each SAFW train can be actuated and controlled from the control room.</p>	<p>24 months</p>

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3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LC0 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST water volume not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours
	<u>AND</u> A.2 Restore CST water volume to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify the CST water volume is \geq 22,500 gal.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains, ~~two CCW heat exchangers~~, and the CCW loop header shall be OPERABLE.

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APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW train inoperable.	A.1 Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met One CCW heat exchanger inoperable.	B.1 Be in MODE 3 Restore CCW heat exchanger to OPERABLE status. AND B.2 Be in MODE 5.	31 days

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CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>6 hours</p> <p>36 hours</p> <p>Required Action and associated Completion Time of Condition A or B not met.</p>	<p>C. Two CCW trains or loop header inoperable. Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 5</p>	<p>C. 6 hours</p> <p>36 hours</p> <p>Initiate Action to restore one CCW train or loop header to OPERABLE status.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.3 Be in MODE 4.</p>

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(continued)

~~Immediately~~

~~6 hours~~

~~12 hours~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Two CCW trains, two CCW heat exchangers, or loop header inoperable.</p>	<p style="text-align: center;">(100)</p> <p>-----NOTE----- LCO 3.0.3 and all other LCO Required Actions are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status.</p> <p>-----</p> <p>D.1 Initiate Action to restore one CCW train, one heat exchanger, and loop header to OPERABLE status.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.3 Be in MODE 4.</p>	<p style="text-align: right;"><i>regaining MODE change</i></p> <p>Immediately</p> <p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.7.1 -----NOTE----- Isolation of CCW flow to individual components does not render the CCW loop header inoperable. -----</p> <p>184 135</p> <p>Verify each CCW manual and power operated valve in the CCW train and heat exchanger flow path and loop header servicing post-accident related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.7.2 Perform a complete cycle of each motor operated isolation valve to the residual heat removal heat exchangers.</p>	<p>In accordance with the Inservice Testing Program</p>



3.7 PLANT SYSTEMS

3.7.8 Service Water (SW) System

LCO 3.7.8 Two SW trains and the SW loop header shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SW train inoperable.	A.1 Restore SW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>ec. Two SW trains or loop header inoperable.</p>	<p>C.1 -----Initiate Action to restore one----- NOTE</p> <p>135</p> <p>Enter applicable conditions and Required Actions of LCO 3.7.7, "CCW System," for the component cooling water heat exchanger(s) made inoperable by SW train or loop header to OPERABLE status.</p> <p>-----</p> <p>AND</p> <p>Enter LCO 3.0.3.2 Be in MODE 3.</p> <p>AND</p> <p>C.3 Be in MODE 4.</p>	<p>Immediately</p> <p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
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31 days SR 3.7.8.2

NOTE

SR 3.7.8.2

135

Isolation of SW flow to individual components does not render the SW loop header inoperable.

Verify each SW manual, power operated, and automatic valve in the SW train flow path and loop header that is not locked, sealed, or otherwise secured in position, is in the correct position.

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~~24 months~~ SR 3.7.8.3

Verify all SW loop header cross-tie valves
are locked in the correct position.

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SR-3.7.8.3

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SR 3.7.8.4	Verify each SW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
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SR 3.7.8.5	Verify each SW pump starts automatically on an actual or simulated actuation signal.	24 months
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3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Air Treatment System (CREATS)

LCO 3.7.9 The CREATS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CREATS filtration train inoperable.	A.1 Restore CREATS filtration train to OPERABLE status.	48 hours
	<p><u>OR</u></p> <p>A.2 -----NOTE----- The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition. -----</p> <p>Place isolation dampers in CREATS Mode F.</p>	48 hours
<p>B. -----NOTE----- Separate Condition entry allowed for each damper. -----</p> <p>One CREATS isolation damper in one or more outside air flowpaths inoperable.</p>	B.1 Restore isolation damper to OPERABLE status.	7 days



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)		
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A or B not met in MODE 5 or 6 or during movement of irradiated fuel.	D.1 Place OPERABLE isolation damper(s) in CREATS Mode F. <u>AND/OR</u> D.22.1 Suspend CORE ALTERATIONS. <u>AND</u> D.32.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately
E. Two CREATS isolation dampers for one or more outside air supply flow paths inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>135</p> <p>169</p> <p>F. Two CREATS isolation dampers for one or more outside air supply flow paths inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies, or during CORE ALTERATIONS.</p>	<p>F.1 Initiate actions to restore one isolation damper to OPERABLE status.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>F.2 Suspend CORE ALTERATIONS.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>F.3 Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate the CREATS filtration train \geq 15 minutes.	31 days
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.9.3	Verify the CREATS actuates on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.10 Auxiliary Building Ventilation System (ABVS)

LCO 3.7.10 The ABVS shall be OPERABLE and in operation.

APPLICABILITY: During movement of irradiated fuel assemblies in the Auxiliary Building when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The ABVS is inoperable.	<p>A.1 ---Suspend movement of irradiated fuel assemblies in the Auxiliary Building.</p> <p>NOTE: LCO 3.0.3 is not applicable.</p> <p>-----</p> <p>Suspend movement of irradiated fuel assemblies in the Auxiliary Building.</p>	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.10.1 Verify ABVS is in operation.	24 hours

SURVEILLANCE	FREQUENCY
<p>SR 3.7.10.2 Perform required Spent Fuel Pool Charcoal Adsorber System filter testing in accordance with the Ventilation Filter Testing Program (VFTP) respect to the outside environment (at the Auxiliary Building operating floor) level.</p> <p>144</p> <p>at</p>	<p>24 hours</p>
<p>(continued)</p>	
<p>SR 3.7.10.3 Perform required Spent Fuel Pool Charcoal Adsorber System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).</p>	<p>In accordance with the VFTP</p>

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool (SFP) Water Level

LCO 3.7.11 The SFP water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFP water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the SFP.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify the SFP water level is \geq 23 ft above the top of the irradiated fuel assemblies seated in the storage racks. (135)	317 days

3.7 PLANT SYSTEMS

3.7.12 Spent Fuel Pool (SFP) Boron Concentration

LCO 3.7.12 The SFP boron concentration shall be within the limit specified in the COLR ~~300 ppm~~.

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APPLICABILITY: When fuel assemblies are stored in the SFP and a SFP verification has not been performed since the last movement of fuel assemblies in the SFP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. SFP boron concentration not within limit.</p>	<p>-----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>A.1 Suspend movement of fuel assemblies in the SFP.</p> <p><u>AND</u></p> <p>A.2.1 Initiate action to restore SFP boron concentration to within limit.</p> <p><u>OR</u></p> <p>A.2.2 Initiate action to perform SFP verification.</p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.12.1 Verify the SFP pool boron concentration is within the limit specified in the COLR.	31 days

107
220

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Storage

LCO 3.7.13 Fuel assembly storage in the spent fuel pool shall be maintained as follows:

147

- a. Fuel assemblies in Region 1 shall have a K-infinity of ≤ 1.458 ~~in the normal reactor core configuration and cold conditions~~; and
- b. Fuel assemblies in Region 2 shall have initial enrichment and burnup within the acceptable area of the Figure 3.7.13-1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met for either region.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Initiate action to move the noncomplying fuel assembly from the applicable region.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.13.1 -----NOTE----- Not required to be performed when transferring a fuel assembly from Region 2 to Region 1. -----</p> <p>Verify by administrative means the K-infinity of the fuel assembly is ≤ 1.458 in the normal reactor core configuration and cold conditions.</p> <p>(147)</p>	<p>Prior to storing the fuel assembly in Region 1</p>
<p>SR 3.7.13.2 Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-1.</p>	<p>Prior to storing the fuel assembly in Region 2</p>

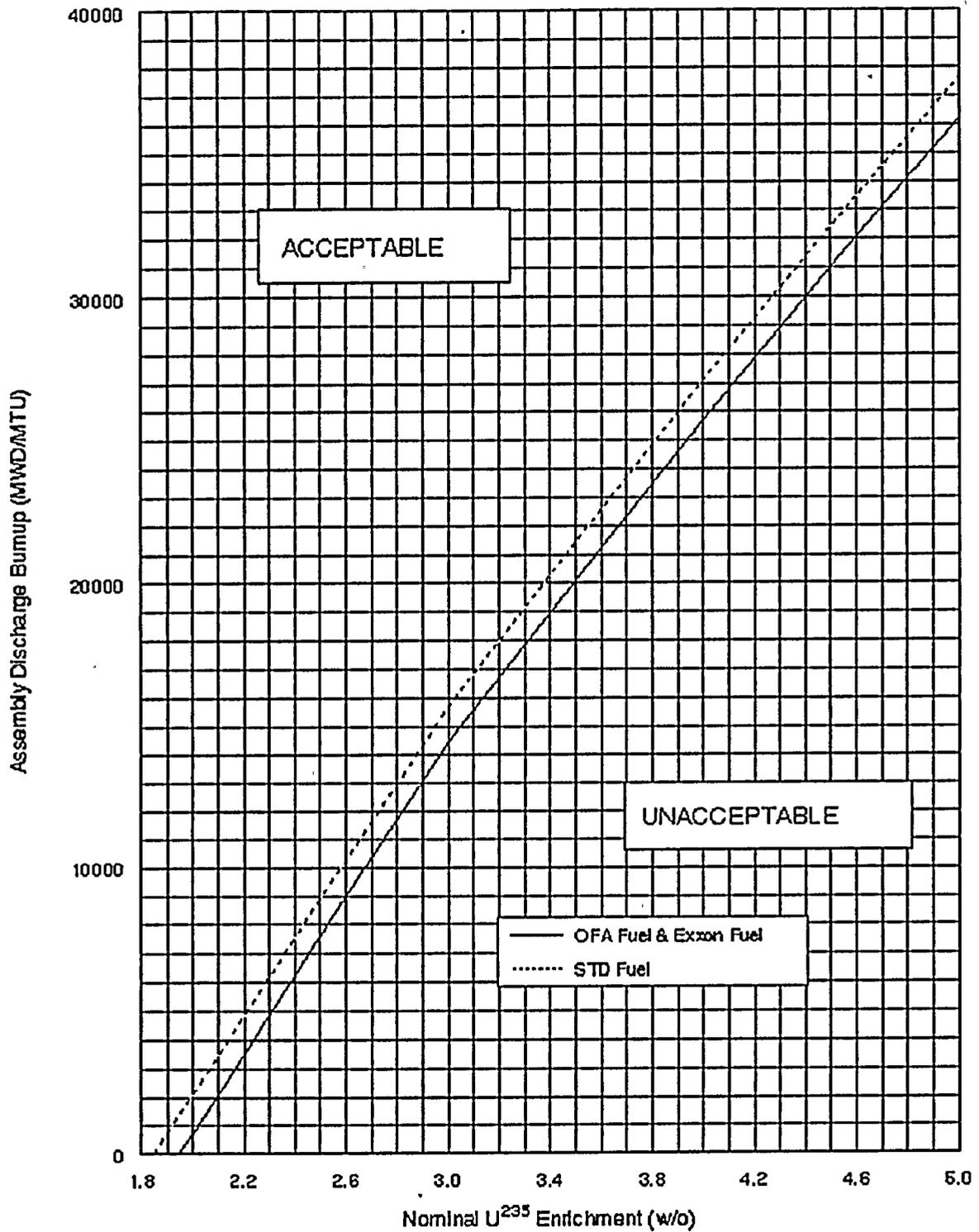


Figure 3.7.13-1
Fuel Assembly Burnup Limits in Region 2

3.7 PLANT SYSTEMS

3.7.14 Secondary Specific Activity

LCO 3.7.14 The specific activity of the secondary coolant shall be
 $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit. 135	A.1 Be in MODE 3.	86 hours
	AND A.2 Be in MODE 5.	4036 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the specific activity of the secondary coolant is $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$.	31 days

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred (but non safety related) heat sink, provided by the condenser and circulating water system, is not available.

Four MSSVs are located on each main steam header, outside containment in the Intermediate Building, upstream of the main steam isolation valves (Ref. 1). MSSVs 3509, 3511, 3513, and 3515 are located on the steam generator (SG) A main steam header while MSSVs 3508, 3510, 3512 and 3514 are located on the SG B main steam header. The MSSVs are designed to limit the secondary system to $\leq 110\%$ of design pressure when passing 100% of design flow. The MSSV design includes staggered setpoints so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine/reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs is to limit the secondary system pressure to $\leq 110\%$ of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

Q3

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased RCS heat removal events (Ref. 2). Of these, the full power loss of external load event is the limiting DBAAOO. This event also results in the loss of normal feedwater flow to the SGs.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The transient response for a loss of external load event without a direct reactor trip (i.e., loss of load when < 50% RTP) presents no hazard to the integrity of the RCS or the Main Steam System. For transients at power levels > 50%, the effect on RCS safety limits is evaluated with no credit taken for the pressure relieving capability of pressurizer spray, the steam dump system, and the SG atmospheric relief valves. The reactor is tripped on high pressurizer pressure with the pressurizer safety valves and MSSVs required to be opened to maintain the RCS and Main Steam System within 110% of their design values.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening (as an initiating event only), and failure to reclose once opened. The passive failure mode is failure to open upon demand which is not considered in the accident analyses.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve SG overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to SR 3.7.1.1 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or secondary system.

(continued)



BASES

APPLICABILITY

In MODES 1, 2, and 3, four MSSVs per SG are required to be OPERABLE to ensure that the RCS remains within its pressure safety limit and that the secondary system, from the SGs to the main steam isolation valves, is limited to $\leq 110\%$ of design pressure for all DBAs.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The SGs are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, the assumptions used in the accident analysis for loss of external load may no longer be valid and the safety valve(s) must be restored to OPERABLE status within 4 hours. This Condition specifically addresses the appropriate ACTIONS to be taken in the event that a non-significant discrepancy related to the MSSVs is discovered with the plant operating in MODES 1, 2, or 3. Examples of this type of discrepancy include administrative (e.g., documentation of inspection results) or similar deviations which do not result in a loss of MSSV capability to relieve steam. The 4 hour Completion Time allows a reasonable period of time for correction of administrative only problems or for the plant to contact the NRC to discuss appropriate action. The 4 hour Completion time is based on engineering judgement.

This Condition is not applicable to a situation in which the ability of a MSSV to open or reclose is questionable. In this event, this Condition is no longer applicable and Condition B of this LCO should be entered immediately since no corrective actions can be implemented during MODES 1, 2, and 3.

(continued)

BASES

ACTIONS
(continued)B.1 and B.2

If the MSSV(s) cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 3), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 4). According to Reference 4, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Setpoint pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria; and
- e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. This SR allows a +1% and -3% setpoint tolerance for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.2.4.
 2. UFSAR, Section 15.2.
 3. ASME, Boiler and Pressure Vessel Code, Section XI.
 4. ANSI/ASME OM-1-1987.
-



B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

BASES

BACKGROUND

The MSIVs (3516 and 3517) isolate steam flow from the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). MSIV closure is necessary to isolate a SG affected by a steam generator tube rupture (SGTR) event or a steam line break (SLB) to stop the loss of SG inventory and to protect the integrity of the unaffected SG for decay heat removal. The MSIVs are air operated swing disk check valves that are held open by an air operator against spring pressure. The MSIVs are installed to use steam flow to assist in the closure of the valve (Ref. 1).

A MSIV is located in each main steam line header outside containment in the Intermediate Building. The MSIVs are downstream from the main steam safety valves (MSSVs) and turbine driven auxiliary feedwater (AFW) pump steam supply, to assure the MSSVs prevent overpressure on the secondary side and assure steam is available to the AFW system following MSIV closure. Closing the MSIVs isolates each SG from the other, and isolates the turbine, steam dump system, and other auxiliary steam supplies from the SGs.

The MSIVs close on a main steam isolation signal generated by either high containment pressure, high steam flow coincident with low T_{avg} and safety injection (SI), or high-high steam flow coincident with SI.

The MSIVs are designed to work with non-return check valves (3518 and 3519) located immediately downstream of each MSIV to preclude the blowdown of more than one SG following a SLB. The MSIVs fail closed on loss of control or actuation power and loss of instrument air once the air is bled off from the supply line. The MSIVs may also be actuated manually.

Each MSIV has a normally closed manual MSIV bypass valve.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The design basis of the MSIVs and non-return check valves is established by the large SLB (Ref. 2). The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity. The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MSIV on the intact SG for this case is assumed to close to prevent excessive cooldown of the RCS which could result in a lower DNB ratio.

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. Due to the non-return check valve on the faulted SG, reverse flow from the steam headers downstream of the MSIV and from the intact SG is prevented from contributing to the energy and mass released inside containment by the SLB. This check valve is a passive device which is not assumed to fail.

SLBs outside of containment can occur in the Intermediate Building and downstream of the MSIVs in the Turbine Building. A SLB in piping > 6 inches diameter in the Intermediate Building is not required to be considered due to an augmented piping inspection program (Ref. 3). For a SLB in the Turbine building, the MSIVs on both SGs must close to isolate the break and terminate the event.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The MSIVs are also credited in a SGTR to manually isolate the SG with the ruptured tube. In addition to minimizing the radiological releases, this assists the operator in isolating the RCS flow through the ruptured SG by preventing the SG from continuing to depressurize and creating a higher pressure difference between the secondary system and the primary system.

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The MSIVs are also considered in other DBAs such as the feedwater line break in which closure of the MSIV on the intact SG maximizes the effect of the break since the energy removal capability of the intact SG would be reduced with respect to long term heat removal.

122

In addition to providing isolation of a faulted SG during a SLB, feedwater line break, or a SGTR, the MSIVs also serve as a containment isolation barrier boundary. The MSIVs are the second containment isolation barrier boundary for the main steam line penetrations which use the steam lines and SGs inside containment as the first barrier boundary. The MSIVs do not receive an automatic containment isolation signal since a spurious signal could result in a significant plant transient.

The MSIVs and non-return check valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

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This LCO requires that two MSIVs and the non-return check valves in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when ~~they are closed and deactivated or when their isolation times are within limits and they can close on an isolation actuation signal.~~ A MSIV must also be capable of isolating a SG for containment isolation purposes. The non-return check valves are considered OPERABLE when they are capable of closing.

This LCO provides assurance that the MSIVs and non-return check valves will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

(continued)

BASES

APPLICABILITY

134

The MSIVs and non-return check valves must be OPERABLE in MODES 1, 2, and 3 when there is significant mass and energy in the RCS and SGs to challenge the integrity of containment, or allow a transient to approach DNBR limits. When the MSIVs are closed and de-activated, and the non-return check valves are closed in MODES 2 and 3, they are already performing their safety function and the MSIVs and their associated non-return check valves are not required to be OPERABLE per this LGO.

In MODE 4, the MSIVs and non-return check valves are normally closed, and the RCS, including and SG energy, is low.

In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and non-return check valves are not required for isolation of potential main steam pipe breaks in these MODES.

ACTIONS.

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The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1-and-A

With one or more valves inoperable in flow path from a SG in MODE 1, action must be taken to restore OPERABLE status within 8 hours. 2

With one MSIV inoperable, action must be taken to restore OPERABLE status or place the MSIV in the closed position within 24 hours. Some repairs to these valves can be made with the plant under hot conditions. Some repairs to the MSIV can be made with the 8 hour Completion Time is reasonable, considering the plant under hot conditions low probability of an accident occurring during this time period that would require a closure of the MSIVs and non-return check valves and the ability to isolate the affected SG by turbine stop valves. —

The 248 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of greater than that

(continued).



BASES

134

~~normally allowed for containment isolation boundaries because the MSIVs and the ability to isolate the affected SG by turbine stop valves that isolate a closed system penetrating containment.~~

~~The 24 hour Completion Time is greater than that normally allowed for containment isolation barriers because the MSIVs are these valves that isolate a differ from most other containment isolation boundaries in that the closed system penetrating containment provides an additional means for containment isolation. These valves differ from most other containment isolation barriers in that the closed system provides an additional means for containment isolation. Failure of this closed system can only result from a SGTR which is not postulated to occur with any other DBA (e.g., LOCA).~~

(continued)

BASES

(continued)

BASES

ACTIONS
(continued)

~~AB.1 and A~~

If the MSIV and/or non-return check valve from a SG cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. 2-
(continued)

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~~For an inoperable MSIV that cannot be restored to OPERABLE to achieve this status within the specified Completion Time, but is closed, the inoperable MSIV, the plant must be verified on a periodic basis to be closed placed in MODE 2 within 6 hours and Condition C would be entered. This Completion Time is necessary reasonable, based on operating experience, to ensure that the assumptions reach MODE 2 in the safety analysis remain valid in an orderly manner without challenging plant systems.—~~

The 31 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

B1 and C.1

With one?

~~Since the MSIVs and non-return check valve inoperable are required to be OPERABLE in MODES 2 and 3, action must be taken to restore the inoperable valve(s) may either be restored to OPERABLE status or isolate the affected main steam line (the associated MSIV closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis and the non-return check valve is no longer required.~~

~~The 8 hour Completion Time is consistent with inoperable valve that allowed in Condition A.—~~

~~A check valve is a passive device For inoperable valves that cannot be inspected or maintained under hot conditions restored to OPERABLE status within the specified Completion Time, but the associated MSIV is closed, the MSIV~~

(continued)

BASES

134

~~must be verified on a periodic basis to be closed.-
Therefore, the inoperability of the non-return check valve
will most likely result from non-significant Program
discrepancies. The 24-hour Completion Time allows a
reasonable period of time to correct the discrepancy. The
24-hour Completion Time is based on engineering judgement
and the installed in-series MSIVs.~~

C.1 and C.2

~~If the MSIV cannot be restored to OPERABLE status or is not
closed and periodically verified closed within the
associated Completion Time, or the non-return check valve is
not restored to OPERABLE status or closed within the
associated Completion Time, the plant must be placed in a
MODE in which the LCO does not apply. To achieve this
status, the plant must be placed at least in MODE 3 within
6 hours, and in MODE 4 within 12 hours. The allowed
Completion Times are reasonable, based on operating
experience, to reach the required plant conditions from full
power conditions in an orderly manner and without
challenging plant systems.~~

(continued)



BASES

134
~~The 24-hour Completion Time allows a reasonable period of time~~ This is necessary to correct the discrepancy ensure that the assumptions in the safety analysis remain valid. The 24-hour 7 day Completion Time is reasonable, based on engineering judgement, in view of MSIV status indications available in the control room, and the installed in-series MSIV other administrative controls, to ensure that these valves are in the closed position.

(continued)

BASES

ACTIONS
~~(continued)~~

D.1

~~If one or more valves in the flowpath from each SG are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately and D. This Condition must be entered when any combination of:~~

(continued)

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If the MSIVs and/or non-return check valves are inoperable such that at least one valve cannot be restored to OPERABLE status or the associated MSIV is inoperable in each of the two main steam flowpaths not closed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 conditions in an orderly manner without challenging plant systems.

E.1

If one or more valves in the flow path from each SG are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MSIVs and non-return check valves are inoperable such that at least one valve is inoperable in each of the two main steam flow paths.

SURVEILLANCE
REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds under no flow and no load conditions. The MSIVs are swing-disk check valves that are held open by their air operators against spring pressure. Once the MSIVs begin to close during hot conditions, the steam flow will assist the valve closure such that testing under no flow and no load conditions is conservative. The 5 second closure time is consistent with the expected response time for

(continued)

BASES

instrumentation associated with the MSIV and the accident analysis assumptions.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.

(continued)

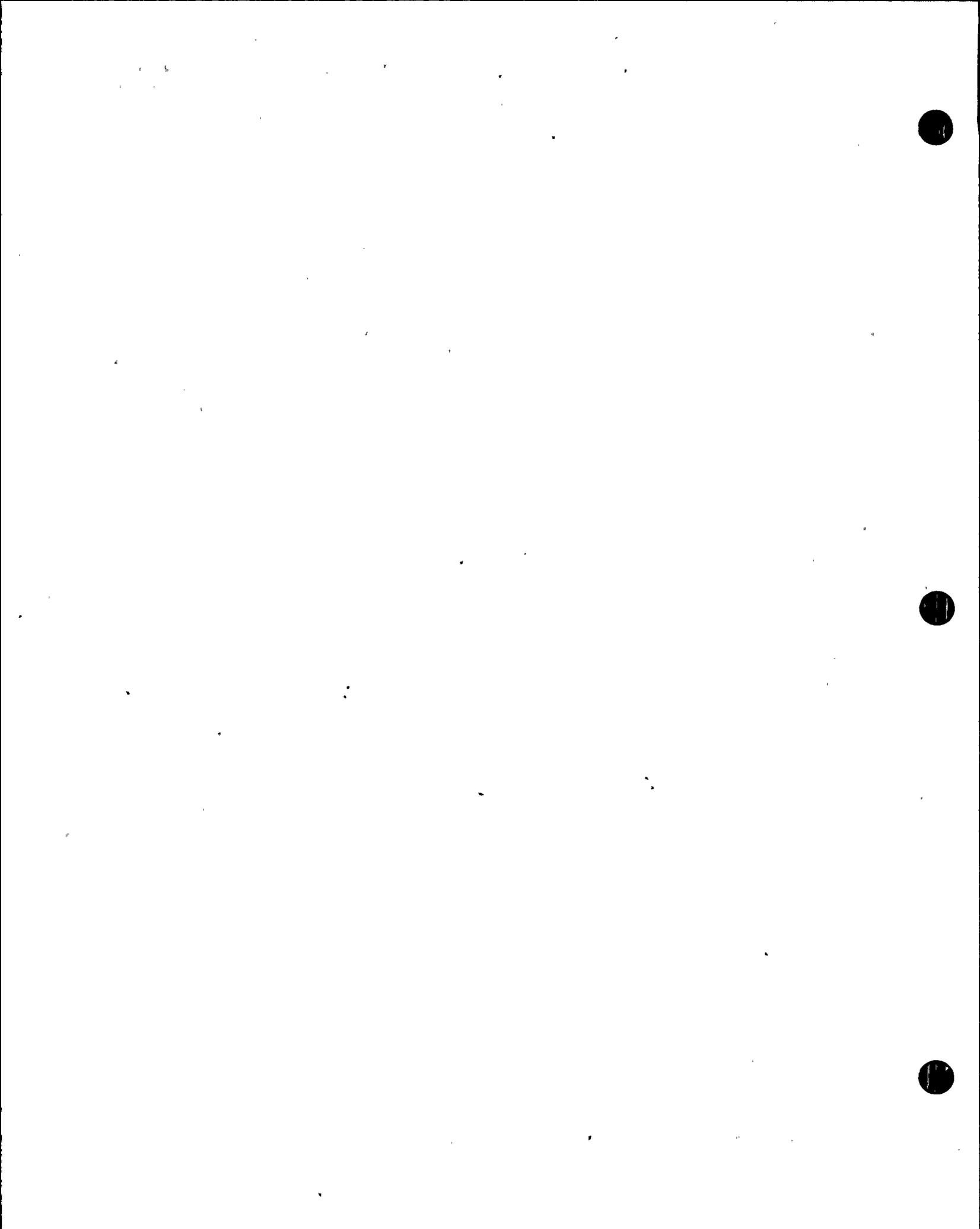
BASES

SURVEILLANCE SR 3.7.2.2

(continued)

REQUIREMENTS

This SR verifies that each main steam non-return check valve can close. As the non-return check valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.



BASES

SURVEILLANCE — SR 3.7.2.3

REQUIREMENTS

(continued) — This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MSIVs should not be tested at power, since even a partial stroke exercise increases the risk of a valve closure and plant transient when the plant is above MODE 4. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODES 1, 2 and 3.

The frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.4.4.
 2. UFSAR, Section 15.1.5.
 3. UFSAR, Section 3.6.2.5.1.
 4. 10 CFR 100.11.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
-



B 3.7 PLANT SYSTEMS

95 B 3.7.3 ~~Main Feedwater Pump Discharge Valves (MFPDVs), Main Feedwater~~
~~Regulating Valves (MFRVs), and Associated Bypass Valves~~ ~~Associated~~
~~Bypass Valves,~~
~~and Main Feedwater Pump Discharge Valves (MFPDVs)~~

BASES

BACKGROUND

The ~~MFPDVs (3977 and 3976), MFRVs (4269 and 4270) and their~~
~~associated bypass valves (4271 and 4272), and MFPDVs (3977~~
~~and 3976)~~ isolate main feedwater (MFW) flow to the secondary
side of the steam generators (SGs) following a Design Basis
Accident (DBA). The safety related function of the ~~MFPDVs,~~
~~MFRVs, and associated bypass valves, and MFPDVs~~ is to provide
for isolation of MFW flow to the secondary side of the SGs
terminating the DBA for line breaks occurring downstream of
the valves. Closure effectively terminates the addition of
feedwater to an affected SG, limiting the mass and energy
release for steam line breaks (SLBs) or feedwater line
breaks (FWLBs) inside containment, and reducing the cooldown
effects for SLBs.

136 The ~~MFPDVs, MFRVs, and associated bypass valves, and MFPDVs~~
in conjunction with check valves located downstream of the
isolation valves also provide a pressure boundary for the
controlled addition of auxiliary feedwater (AFW) to the
intact SG (see Figure B 3.7.3-1).

One MFPDV is located in the Turbine Building on the
discharge line of each MFW pump (Ref. 1). One MFRV and
associated bypass valve is located on each MFW line to its
respective SG, outside containment in the Turbine Building.
The ~~MFPDVs, MFRVs, MFRVs, and bypass valves associated bypass~~
~~valves, and MFPVs~~ are located upstream of the AFW injection
point so that AFW may be supplied to the SGs following
closure of the MFRVs and bypass valves. The piping volume
from these valves to the SGs is accounted for in calculating
mass and energy releases, and must be refilled prior to AFW
reaching the SG following either an SLB or FWLB.

(continued)

BASES

BACKGROUND
(continued)

169

The MFPDV closes on the opening of the MFW pump breaker which occurs on receipt of a safety injection signal or a ~~low pump suction pressure~~ any other signal which trips the pump breaker. The MFRVs and bypass valves close on receipt of a safety injection signal, a SG high level signal, or on a reactor trip with $T_{avg} < 554^{\circ}F$ with the associated MFRV in auto. All valves may also be actuated manually. In addition to the ~~MFPDVs, MFRVs, and associated bypass valves and MFPDVs~~, a check valve located outside containment for each feedwater line is available. The check valve isolates the feedwater line penetrating containment providing a containment isolation ~~barrier~~ boundary.

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APPLICABLE
SAFETY ANALYSES

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The design basis of the ~~MFPDVs, MFRVs, and associated bypass valves is, and~~ MFPDVs is established by the analyses for the SLB. The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity (Ref. 2). The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MFRV and bypass valve on the intact SG for this case are assumed to close on a safety injection signal to prevent excessive cooldown of the RCS which could result in a lower DNB ratio. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps, which have their breakers opened by a SI signal, and the MFPDV which close on opening of the MFW pump breakers.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant, maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. The MFRV and bypass valve on the faulted SG are assumed to close on a safety injection signal to prevent continued contribution to the energy and mass released inside containment by the SLB. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps and closure of the MFPDVs.

The MFRVs and bypass valves are also credited for isolation in the feedwater transient analyses (e.g., increase in feedwater flow). These valves close on either a safety injection or high SG level signal depending on the scenario. The valves also must close on a FWLB to limit the amount of additional mass and energy delivered to the SGs and containment.

The failure of the MFRVs to control flow is also considered as an initiating event. This includes consideration of a valve failure coincident with an atmospheric relief valve failure since a single component in the Advanced Digital Feedwater Control System (ADFCS) controls both components (Ref. 3). This combined valve failure accident scenario is evaluated with respect to DNB since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the SLB accident.

95 The MFPDVs, MFRVs, and associated bypass valves and MFPDVs satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

95

This LCO ensures that the MFPDVs, MFRVs, and associated bypass valves, and MFPDVs will isolate MFW flow to the SGs, following a FWLB or SLB.

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This LCO requires that two MFPDVs, two MFRVs, and two MFRV bypass valves be OPERABLE. The MFPDVs, MFRVs, and associated bypass valves, and MFPDVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal ~~or when the valves are closed and de-activated, or isolated by a closed manual valve.~~

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. It may also result in the introduction of water into the main steam lines for an excess feedwater flow event.

APPLICABILITY

95

The MFPDVs, MFRVs, and associated bypass, and MFPDVs valves must be OPERABLE whenever there is significant mass and energy in the RCS and SGs. This ensures that, in the event of a DBA, the accident analysis assumptions are maintained. In MODES 1, 2, and 3, the MFPDVs, MFRVs, and the associated bypass valves, and MFPDVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve such that both SGs are isolated from both MFW pumps, they are already performing their safety function and no longer required to be OPERABLE.

In MODE 4, the MFPDVs, MFRVs, and associated bypass valves, and MFPDVs are normally closed since AFW is providing decay heat removal due to the low SG energy level. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MFPDVs, MFRVs, and associated bypass valves, and MFPDVs are not required for isolation of potential pipe breaks in these MODES.

(continued)

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one or more MFPDV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or close the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFPDV that is closed must be verified on a periodic basis that it remains closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 317 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

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B.1 and B.2

With one or more MFRV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

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An inoperable MFRV that is closed must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 317 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

135
With one or more MFRV bypass valve(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV bypass valve that is closed must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 31 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

D.1 and D.2

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If the MFPDV, MFRV, or associated bypass valve, or MFPDV cannot be restored to OPERABLE status or closed within 24 hours or cannot be verified closed once per 31 days, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

2000



2000

BASES

ACTIONS
(continued)

E.1

169

If one or more MFPDV(s) and one or more MFRV(s), or one or more MFPDV(s) and one or more MFPDV/MFRV bypass valve(s) are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MFPDVs, MFRVs, and associated bypass valves or MFPDVs are inoperable such that a MFW pump, condensate pump, or condensate booster pump can provide unisolable flow to one or both SGs (see Figure B 3.7.3-1).

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SURVEILLANCE
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFPDV is ≤ 80 seconds from the full open position on an actual or simulated actuation signal (i.e., from opening of MFW pump breakers). The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI, (Ref. 4) requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

135

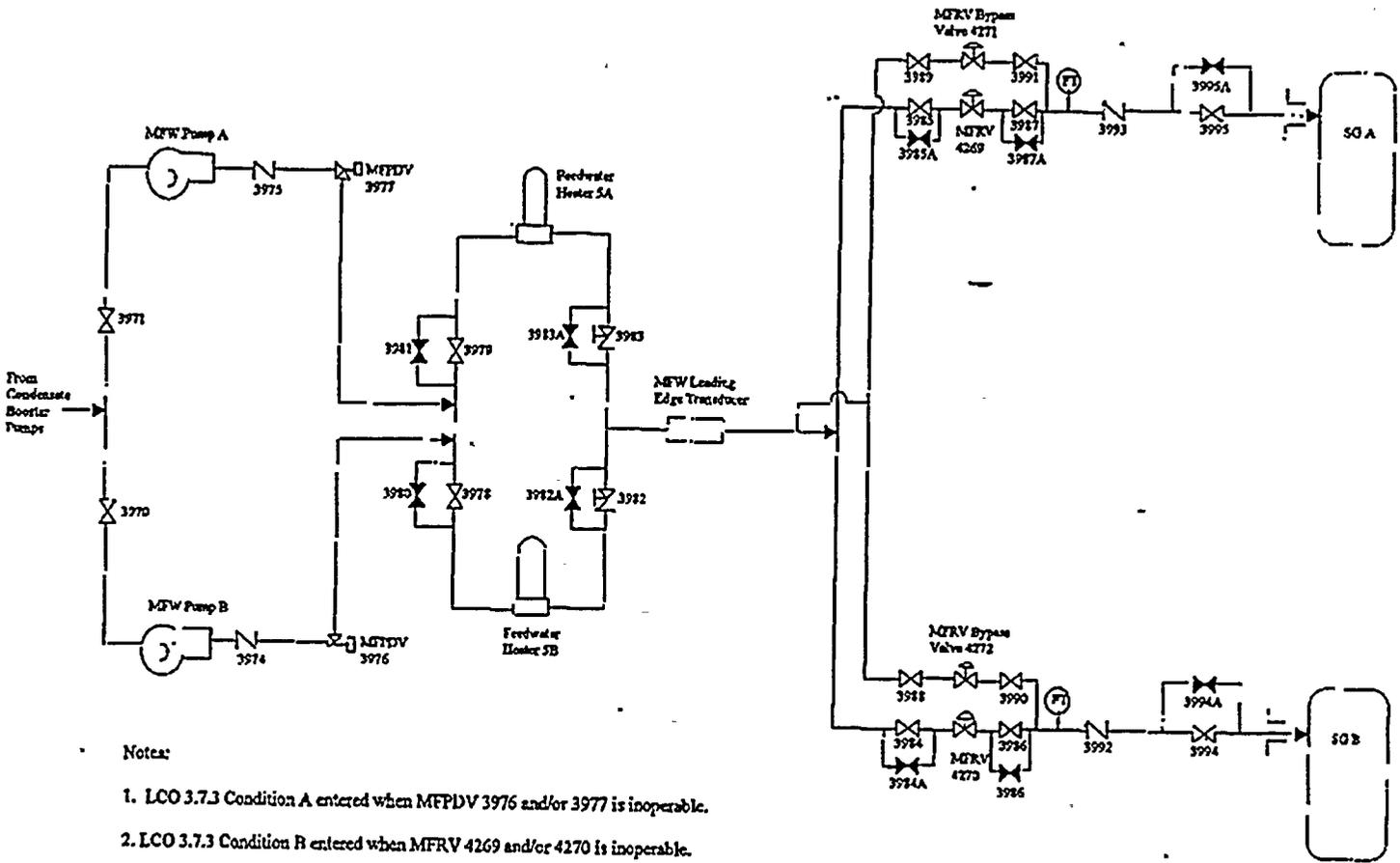
SR 3.7.3.2

This SR verifies that the closure time of each MFRV and associated bypass valve is ≤ 10 seconds from the full open position on an actual or simulated actuation signal. The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 4), requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 10.4.5.3.
 2. UFSAR, Section 15.1.5..
 3. UFSAR, Section 15.1.6.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
-



Notes:

1. LCO 3.7.3 Condition A entered when MFPDV 3976 and/or 3977 is inoperable.
2. LCO 3.7.3 Condition B entered when MFRV 4269 and/or 4270 is inoperable.
3. LCO 3.7.3 Condition C entered when MFRV Bypass Valve 4271 and/or 4272 is inoperable.
4. LCO 3.7.3 Condition E entered when any combination of valve inoperabilities results in an unsalvageable flowpath from the condensate booster pumps to one or more SGs.

For illustration only

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Figure B 3.7.3-1
 MFRVs, Associated Bypass Valves and MFPDVs

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Relief Valves (ARVs)

BASES

(93)

BACKGROUND

There is an ARV (3410 and 3411) located on the main steam header from each steam generator (SG). The ARVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ARVs have two functions (Ref. 1):

- a. provide secondary system overpressure protection below the setpoint of the main steam safety valves (MSSVs); and
- b. provide a method for cooling the plant should the preferred heat sink via the steam dump system to the condenser not be available.

The accident analyses do not credit either of these functions since the ARVs do not have a safety-related source of motive air and the accident analyses do not typically require cooldown to the residual heat removal entry conditions since the plant was originally designed to maintain Hot Shutdown conditions indefinitely. The only exception is with respect to steam generator tube rupture (SGTR) events which require the use of at least one ARV to provide heat removal from the Reactor Coolant System (RCS) to prevent saturation conditions from developing.

(169)

The ARVs are air operated valves located in the Intermediate Building with a relief capacity of 329,000 lbm/hr each (approximately 5% of RTP power). The ARVs are normally closed, fail closed valves which receive motive air from the instrument air system. The valves can also receive motive air from a non-seismic backup nitrogen bottle bank system. The valves are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are normally controlled by the Advanced Digital Feedwater Control System (ADFCS) but can also be remote manually operated and opened locally by use of handwheels located on the valves.

(continued)

BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The design basis for the ARVs is established by the SGTR event (Ref. 2). For this accident scenario, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. Following a SGTR, the MSSVs will maintain the secondary system pressure at approximately 1085 psig which could result in the loss of subcooling margin since the RCS average temperature is attempting to stabilize at approximately 547°F. The ARVs are used during the first 30 to 60 minutes of the SGTR to continue the RCS cooldown in an effort to reduce, and eventually terminate, the primary to secondary system flow in the ruptured SG. The inability to cooldown could result in inadequate subcooling margin which would delay the termination of the leakage through the ruptured tube.

The opening of the ARVs is also considered coincident with a failure of a main feedwater regulating valve (Ref. 3) since a single component in the ADFCS controls both components. This combined valve failure accident scenario is evaluated with respect to departure from nucleate boiling since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the steam line break accident.

93

The ARVs are equipped with block valves in the event the ARV spuriously fails to open or fails to close during use.

The ARVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

93

~~Two ARVs are required to be OPERABLE for and their associated manual operation either locally (using the handwheel or local panel) or remotely to relieve main steam pressure. Block valves are required to be OPERABLE. Failure to meet the LCO can result in the inability to cool the plant following SGTR event in which the condenser is unavailable. The ARVs are required for use with the manual operation either locally (using the handwheel or local panel) or remotely to relieve main steam dump system pressure.~~

~~An~~ The ARV is considered block valves must be OPERABLE when it is capable of being manually opened within 20 minutes of determining the need to utilize the ARV following isolate a

(continued)

BASES

~~SGTR failed open ARV. The A closed block valve does not render it or its ARV must also be capable of closing within 15 minutes in the event the valve spuriously opens online inoperable if operator action time to open the SG with the ruptured tube block valve can be accomplished within the time frames specified below. Finally, Failure to meet the ARV must be capable of closing within 5 minutes. CO can result in the event that the ARV on the intact SG fails to close, inability to cool the plant following initialization of a cool-down SGTR event in which the condenser is unavailable for use with the steam dump system.~~

(continued)

BASES

169
LCO (continued) An ARV line is considered OPERABLE when it is capable of being manually opened within 20 minutes of determining the need to utilize the ARV following a SGTR. The ARV line must also be capable of closing within 15 minutes in the event the ARV spuriously opens on the SG with the ruptured tube. Finally, the ARV line must be capable of closing within 5 minutes in the event that the ARV on the intact SG fails to close following initiation of a cooldown. For the closure requirements, either the ARV or its associated block valve may be credited for OPERABILITY.

43
APPLICABILITY In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, the ARVs/ARV lines are required to be OPERABLE.

In MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODE 4, the ARVs are not required since the saturation pressure of the reactor coolant is below the lift settings of the MSSVs. In MODE 5 or 6, an SGTR is not a credible event since the water in the SGs is below the boiling point and RCS pressure is low.

ACTIONS

A.1

133
With one ARV line inoperable, action must be taken to restore the valve to OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ARV line and a nonsafety grade backup in the steam dump system.

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply since the steam dump system would normally be in service during lower MODES of operation and can provide an acceptable alternative to the inoperable ARV line.

(continued)

BASES

(continued)

ACTIONS B.1

133

If the ARV line cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 with RCS average temperature < 500°F within 8 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both ARVs ARV lines are inoperable, the plant is in a condition outside of the accident analyses for a SGTR event; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.7.4.1

To perform a cooldown of the RCS, the ARVs must be able to be opened either remotely or locally. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

ISR 3.7.4.2

96

The function of the block valve is to isolate a failed open ARV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve

(continued)

BASES

96

during plant cool down may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.



BASES

- REFERENCES
1. UFSAR, Section 10.3.2.5.
 2. UFSAR, Section 15.6.3.
 3. UFSAR, Section 15.1.6.
-



B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System supplies feedwater to the steam generators (SGs) to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the main steam safety valves (MSSVs) or atmospheric relief valves. If the main condenser is available, steam may be released via the steam dump valves. The AFW System is comprised of two separate systems, a preferred AFW System and a Standby AFW (SAFW) System (Ref. 1).

AFW System

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The preferred AFW System consists of two motor driven AFW (MDAFW) pumps and one turbine driven AFW (TDAFW) pump configured into three separate trains which are all located in the Intermediate Building (see Figure B 3.7.5-1). Each motor driven MDAFW train provides 100% of AFW flow capacity, and the turbine driven TDAFW pump provides 200% of the required capacity to the SGs, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to the condensate storage tanks (CSTs). Each motor driven AFW MDAFW train is powered from an independent Class 1E power supply and feeds one SG, although each pump has the capability to be realigned from the control room to feed the other SG via cross-tie lines containing normally closed motor operated valves (4000A and 4000B). The two motor driven AFW MDAFW trains will actuate automatically on a low-low level signal in either SG, opening of the main feedwater (MFW) pump breakers, a safety injection (SI) signal, or the ATWS mitigation system actuation circuitry (AMSAC). The pumps can also be manually started from the control room.

(continued)

BASES (continued)

BACKGROUND

(continued)
(continued)

139

The ~~steam turbine driven AFW~~ pump receives steam from each main steam line upstream of the two main steam isolation valves. Either of the steam lines will supply 100% of the requirements of the ~~turbine driven AFW~~ pump. The ~~turbine driven AFW~~ pump supplies a common header capable of feeding both SGs by use of fail-open, air-operated control valves (4297 and 4298). The ~~turbine driven AFW~~ pump will actuate automatically on a low-low level signal in both SGs, loss of voltage on 4160 V Buses 11A and 11B, or the ATWS mitigation system actuation circuitry (AMSAC). The pump can also be manually started from the control room.

The normal source of water for the AFW System is the CSTs which are located in the non-seismic Service Building. The Service Water (SW) System (LCO 3.7.8) can also be used to supply a safety-related source of water through normally closed motor operated valves (4013, 4027, and 4028) which supply each AFW train.

SAFW System

169

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The SAFW System consists of two motor driven pumps configured into two separate trains (see Figure B 3.7.5-2). Each motor driven SAFW train provides 100% of the AFW flow capacity as assumed in the accident analyses and supplies one SG through the use of a normally open motor-operated stop check valve. Each pump has the capability to be realigned from the control room to feed the other SG via normally closed motor operated valves (9703A and 9703B). Each pump is powered from an independent Class 1E power supply and can be powered from the diesel generators provided that the breaker for the associated ~~AFW~~ pump is opened. The safety-related source of water for the SAFW System is the SW System through two normally closed motor operated valves (9629A and 9629B). Condensate can also be supplied by a 10,000 gallon condensate test tank and the yard fire hydrant yard loop.

The SAFW System is manually actuated in the event that the preferred AFW System has failed due to a high energy line break (HELB) in the Intermediate Building, a seismic or fire event. The SAFW trains are located in the SAFW Pump Building located adjacent to the Auxiliary Building.

(continued)

BASES (continued)

BACKGROUND
(continued)

The SAFW Pump Building environment is controlled by room coolers which are supplied by the same SW header as the pump trains. These coolers are required when the outside air temperature is $\geq 80^{\circ}\text{F}$ to ensure the SAFW Pump Building remains $\leq 120^{\circ}\text{F}$ during accident conditions.

169
The AFW System is designed to supply sufficient water to the SG(s) to remove decay heat with SG pressure at the lowest MSSV set pressure plus 1% of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the plant to RHR entry conditions, with steam released through the ARVs.

APPLICABLE
SAFETY ANALYSES

The design basis of the AFW System is to supply water to the SG(s) to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the SGs at pressures corresponding to the lowest MSSV set pressure plus 1%.

The AFW System mitigates the consequences of any event with the loss of normal feedwater. The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows (Ref. 2):

- a. Feedwater Line Break (FWLB);
- b. Loss of MFW (with and without offsite power);
- c. Steam Line Break (SLB);
- d. Small break loss of coolant accident (LOCA);
- e. Steam generator tube rupture (SGTR); and
- f. External events (tornados and seismic events).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

The AFW System design is such that any of the above DBAs can be mitigated using the preferred AFW System or SAFW System. For the FWLB, SLB, and external events DBAs (items a, c, and f), the worst case scenario is the loss of all three preferred AFW trains due to a HELB in the Intermediate or Turbine Building, or a failure of the Intermediate Building block walls. For these three events, the use of the SAFW System within 10 minutes is assumed by the accident analyses. Since a single failure must also be assumed in addition to the HELB or external event, the capability of the SAFW System to supply flow to an intact SG could be compromised if the SAFW cross-tie is not available. For HELBs within containment, use of either the SAFW System or the AFW System to the intact SG is assumed within 10 minutes.

For the SGTR events (item e), the accident analyses assume that one AFW train is available upon a SI signal or low-low SG level signal. Additional inventory is being added to the ruptured SG as a result of the SGTR such that AFW flow is not a critical feature for this DBA.

For the loss of MFW events and small break LOCA (items b and d), two trains of AFW are assumed available (i.e., two ~~motor driven AFW/DAFW~~ trains or the ~~turbine driven AFW/DAFW~~ train) upon a low-low SG level signal and SI signal, respectively. Two AFW trains are assumed available since no single failure can result in the loss of more than one AFW train. The loss of MFW is a Condition 2 event (Ref. 3) which places limits on the response of the RCS from the transient (e.g., no challenge to the pressurizer power operated relief valves is allowed). Two trains of AFW are required to maintain these limits. The small break LOCA analysis requires two trains of AFW to lower RCS pressure below the shutoff head of the SI pumps.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

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In addition to its accident mitigation function, the energy and mass addition capability of the AFW System is also considered with respect to HELBs within containment. For SLBs and FWLBs within containment, pump runout from all three AFW pumps is assumed for 10 minutes until operations can isolate the flow by tripping the AFW pumps or by closing the respective pump discharge flowpath(s) flow path(s). Therefore, the motor operated discharge isolation valves for the motor operated AFWD/DAFW pump trains (4007 and 4008) are designed to limit flow to < 230 gpm. The TDAFW train is assumed to be at runout conditions (i.e., 600 gpm).

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary or containment.

The AFW System is comprised of two systems which are configured into five trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the SGs are OPERABLE (see Figures B 3.7.5-1 and 3.7.5-2). This requires that the following be OPERABLE:

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is available

- a. Two motor driven AFWD/DAFW trains taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), and capable of supplying their respective SG with ≥ 200 gpm and ≤ 230 gpm total flow;
- b. The turbine AFWD/DAFW train taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), provided steam flow from both main steam lines upstream of the MSIVs, and capable of supplying both SGs with ≥ 200 gpm each; and

(continued)

BASES (continued)

ECO
(continued) c. Two motor driven SAFW trains capable of being-
initiated either locally or from the control room
within 10 minutes, taking suction from the SW System,
and supplying their respective SG and the opposite SG
through the SAFW cross-tie line with ≥ 200 gpm.

ECO
(continued) The piping, valves, instrumentation, and controls in
the
required flow paths are also required to be OPERABLE. The
cross tie line for the preferred AFW motor driven pump TDAFW
train is not required for this LCO comprised of a common pump
and two flow paths. A TDAFW train flow path is defined as
the steam supply line and the SG injection line from/to the
same SG. The failure of the pump or both flow paths renders
the TDAFW train inoperable.

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The cross tie line for the preferred MDAFW pumps is not
required for this LCO. The recirculation lines for the
preferred AFW system and SAFW system pumps are not credited
in the accident analysis and are also not required to be
OPERABLE for this LCO since the MSSVs maintain the SG
pressure below the pump's shutoff head.

The SAFW Pump Building room coolers are required to be
OPERABLE when the outside air temperature is $\geq 80^\circ\text{F}$. If one
room cooler is inoperable, the associated SAFW train is
inoperable.

APPLICABILITY

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In MODES 1, 2, and 3, the AFW System is required to be
OPERABLE in the event that it is called upon to function
when the MFW System is lost. In addition, the AFW System is
required to supply enough makeup water to replace the SG
secondary inventory ~~lost~~ as the plant cools to MODE 4 ^{lost}
conditions.

In MODE 4, 5, or 6, the SGs are not normally used for heat
removal, and the AFW System is not required.

(continued)

BASES (continued)

ACTIONS

A.1

139

~~If one motor driven AFW train is inoperable, or one of the turbine driven AFW train flowpaths, TDAFW train flow paths is inoperable, action must be taken to restore the flowpath or the AFW motor driven train flow path to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:~~

- a. ~~The redundant OPERABLE turbine driven AFW pump flowpath flow path;~~
- b. ~~The availability of redundant OPERABLE motor driven AFW MDAFW and SAFW pumps; and~~

ACTIONS

~~A. c. 1 (continued)~~

~~The low probability of an event occurring that requires the inoperable TDAFW pump flow path.~~

~~The low probability of an event occurring that requires the inoperable turbine driven AFW pump flowpath or motor driven AFW pump A TDAFW train flow path is defined as the steam supply line and SG injection line from/to the same SG.~~

~~A turbine driven AFW train flowpath is defined as the steam supply line and SG injection line from/to the same SG.~~

B1

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~~If one MDAFW train is inoperable, action must be taken to restore the train to OPERABLE status within 7 days. 1~~

~~With The 7 day Completion Time is reasonable, based on the turbine driven AFW train inoperable, both motor driven AFW trains inoperable, or one turbine driven AFW train flowpath and one motor driven AFW train inoperable to opposite SGs, action must be taken to restore OPERABLE status within 72 hours following reasons:~~

- a. ~~If the inoperable motor driven AFW The redundant OPERABLE MDAFW train supplies the same SG as the~~

(continued)

BASES (continued)

~~inoperable turbine driven flowpath, Condition E must be entered.~~

~~h. — A turbine driven AFW train is comprised of the availability of the pump and two flowpaths: redundant OPERABLE TDAFW and SAFW pumps; and~~

~~c. — A turbine driven AFW train. The low probability of an event occurring that requires the inoperable MDAFW train flowpath is defined as the steam supply line and the SG injection line from/to the same SG.~~

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(continued)



BASES (continued)

~~The combination of failures which requires entry into this Condition all result in the loss of one train (or one flowpath) of preferred AFW cooling to each SG such that redundancy is lost.~~ ACTIONS

(continued)

C.1

139
With the TDAFW train inoperable, ^{or} both MDAFW trains inoperable, or one TDAFW train flow path and one MDAFW train inoperable to opposite SGs, action must be taken to restore OPERABLE status within 72 hours. If the inoperable MDAFW train supplies the same SG as the inoperable TDAFW flow path, Condition E must be entered.

The combination of failures which requires entry into this Condition all result in the loss of one train (or one flow path) of preferred AFW cooling to each SG such that redundancy is lost. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

GD.1

With all AFW trains to one or both SGs inoperable, action must be taken to restore at least one train or TDAFW flow path to each affected SG to OPERABLE status within 4 hours.

The combination of failures which require entry into this Condition all result in the loss of preferred AFW cooling to at least one SG. If a SGTR were to occur in this condition, preferred AFW is potentially unavailable to the unaffected SG. If AFW is unavailable to both SGs, the accident analyses for small break LOCAs and loss of MFW would not be met.

(continued)

BASES (continued)

ACTIONS D.1 (continued)

139

The two MDAFW trains of the preferred AFW System are normally used for decay heat removal during low power operations since air operated bypass control valves are installed in each train to better control SG level (see Figure B 3.7.5-1). Since a feedwater transient is more likely during reduced power conditions, 4 hours is provided to restore at least one train of additional preferred AFW before requiring a controlled cooldown. This will also provide time to find a condensate source other than the SW System for the SAFW System if all three AFW trains are inoperable. The 4 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

E.1

With one SAFW train inoperable, action must be taken to restore OPERABLE status within 14 days. This Condition includes the inoperability of one of the two SAFW cross-tie valves which requires declaring the associated SAFW train inoperable (e.g., failure of 9703B would result in declaring SAFW train D inoperable). The 14 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

ACTIONS DE.1

—(continued)—

With both SAFW trains inoperable, action must be taken to restore at least one SAFW train to OPERABLE status within 7 days. This Condition includes the inoperability of the SAFW cross-tie. The 7 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

(continued)

BASES (continued)

ECTIONS G.1

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With all AFW trains and flow paths to one or both SGs inoperable, action must be taken to restore at least one train or flowpath to each affected SG to OPERABLE status within 4 hours. A turbine driven AFW train flowpath is defined as the steam supply line and the SG injection line from/to the same SG.

The combination of failures which require entry into this Condition all result in the loss of preferred AFW cooling to at least one SG. The two motor driven trains of the preferred AFW System are normally used for decay heat removal during low power operations since air operated bypass control valves are installed in each train to better control SG level. Since a feedwater transient is more likely during reduced power conditions, 4 hours is provided to restore at least one train of additional preferred AFW before requiring a controlled cooldown. This will also provide time to find a condensate source other than the SW System for the SAFW System if all three AFW trains are inoperable. The 4 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

(continued)



BASES (continued)

ACTIONS ~~F.1 and F.2~~
(continued)

139

When Required Action A.1, B.1, C.1, D.1, ~~or E.1, or F.1~~ cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

G.H.1

If all three preferred AFW trains and both SAFW trains are inoperable the plant is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one ~~AEW~~ or SAFW train to OPERABLE status.

MDAFW,
TDAFW,

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For the purposes of this Required Action-G, only one TDAFW train flow path and the pump must be restored to exit this Condition.

Required Action H.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one ~~AEW~~ or SAFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW and SAFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Periodically comparing the reference differential pressure and flow of each AFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

(176) This SR is modified by a Note indicating that the SR is only required to be performed ~~met~~ prior to entering MODE 1 for the ~~turbine driven AFW~~ DAFW pump since suitable test conditions may not have ~~not~~ been established.

(93)

(continued)

BASES

This deferral is required because there is ⁱⁿ sufficient steam pressure to perform the test.

Q3

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.7.5.3

Periodically comparing the reference differential pressure and flow of each SAFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. Because it is undesirable to introduce SW into the SGs while they are operating, this testing is performed using the test condensate tank. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

SR 3.7.5.4

This SR verifies that each AFW and SAFW motor operated suction valve from the SW System (4013, 4027, 4028, 9629A, and 9629B), each AFW and SAFW discharge motor operated valve (4007, 4008, 9704A, 9704B, and 9746), and each SAFW cross-tie motor operated valve (9703A and 9703B) can be operated when required. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 4).

SR 3.7.5.5

This SR verifies that AFW can be delivered to the appropriate SG in the event of any accident or transient that generates an actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.5.6

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month Frequency is based on the potential need to perform this Surveillance under the conditions that apply during a plant outage.

(176) This SR is modified by a Note indicating that the SR is only required to be performed ~~at~~ prior to entering MODE 1 for the ~~turbine driven AFW/D~~ AFW pump since suitable test conditions may ~~not~~ have ~~not~~ been established.

(93) ~~This deferral is required because there is insufficient steam pressure to perform the test.~~

SR 3.7.5.7

This SR verifies that the SAFW System can be actuated and controlled from the control room. The SAFW System is assumed to be manually initiated within 10 minutes in the event that the preferred AFW System is inoperable. The Frequency of 24 months is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed at power.

REFERENCES

1. UFSAR, Section 10.5.
 2. UFSAR Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
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Figure B 3.7.5-1
Preferred AFW System

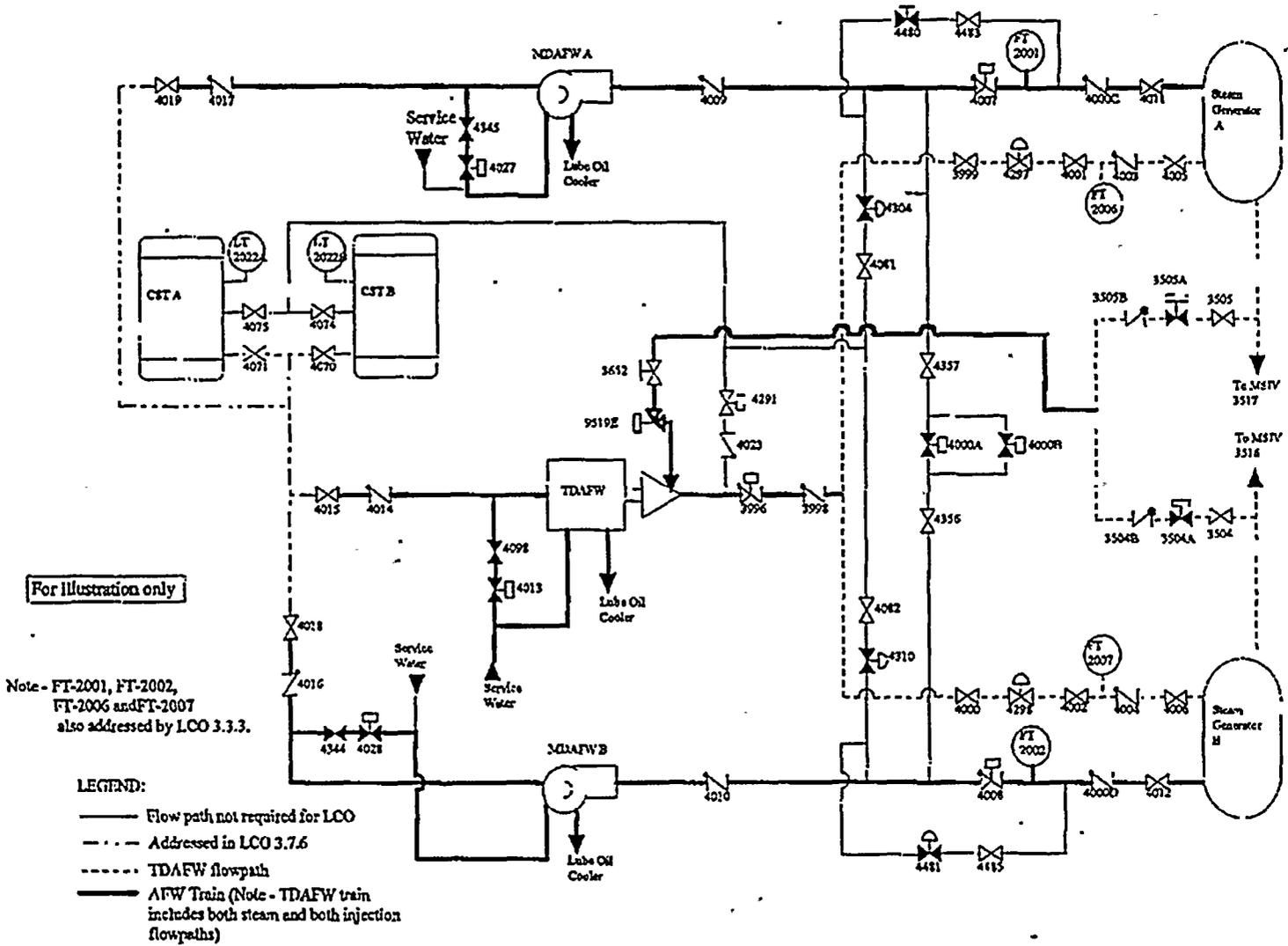
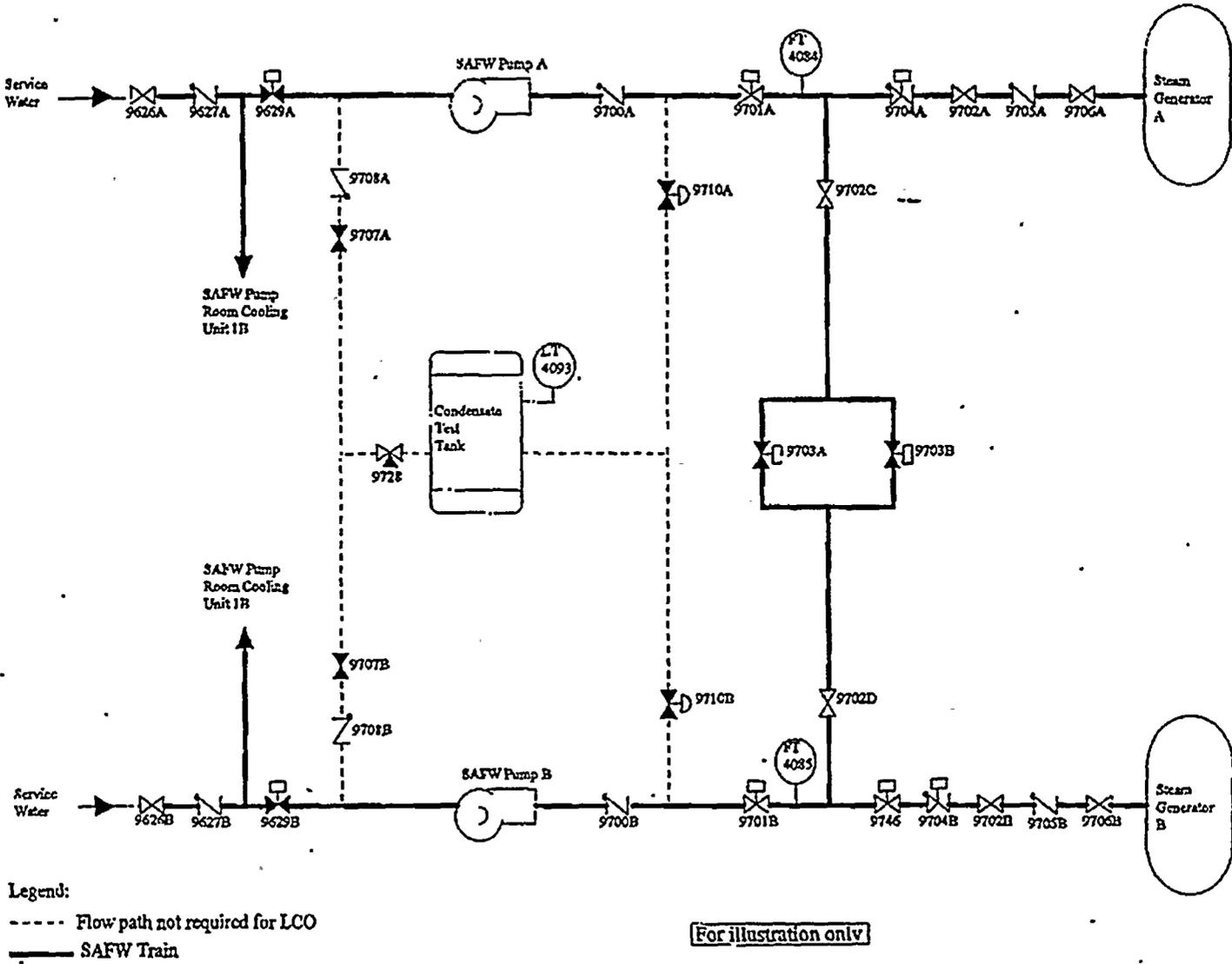


Figure B 3.7.5-2
Standby AFW System



B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tanks (CSTs)

BASES

BACKGROUND

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The CSTs provide a source of water to the steam generators (SGs) for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the preferred Auxiliary Feedwater (AFW) System (LCO 3.7.5) (see Figure B 3.7.5-1). The resulting steam produced in the SGs is released to the atmosphere by the main steam safety valves or the atmospheric relief valves.

When the main steam isolation valves are open, the preferred means of heat removal from the RCS is to discharge steam to the condenser by the nonsafety grade path of the steam dump valves. The condensed steam is then returned to the SGs by the main feedwater system. This has the advantage of conserving condensate while minimizing releases to the environment.

There are two 30,000 gallon CSTs located in the non-seismic Service Building (Ref. 1). The CSTs are not considered safety related components since the tanks are not protected against earthquakes or other natural phenomena, including missiles. The safety related source of condensate for the AFW and Standby AFW Systems is the Service Water (SW) System (LCO 3.7.8). The CSTs are connected by a common header which leads to the suction of all three AFW pumps. A single level transmitter is provided for each CST (LT-2022A and LT-2022B). The CSTs can be refilled from the condenser hotwell or the all-volatile-treatment condensate storage tank.

APPLICABLE SAFETY ANALYSES

The CSTs provide cooling water to remove decay heat and to cooldown the plant following all events in the accident analysis (Ref. 2) which assumes that the preferred AFW System is available immediately following an accident. For any event in which AFW is not required for at least 10 minutes following the accident, the SW System provides the source of cooling water to remove decay heat.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The limiting Design Basis Accident (DBA) for the condensate volume is the loss of normal feedwater event and small break loss of coolant accident (LOCA) (Ref. 2). For the loss of normal feedwater event, flow from at least two AFW pumps is required upon a low level signal in either SG to meet the acceptance criteria for a Condition 2 event (Ref. 3). For the small break LOCA, two AFW pumps are required to lower the RCS pressure below the shutoff head of the safety injection pumps. Assuming that all three AFW pumps initiate at their maximum flowrate, the CSTs provide sufficient inventory for at least 20 minutes (at greater than required flowrates) before operator action to refill the CSTs or transfer suction to the SW System is required.

A nonlimiting event considered in CST inventory determinations is a main feedwater line break inside containment. This break has the potential for dumping condensate until terminated by operator action after 10 minutes since there is no automatic re-configuration of the AFW System. Following termination of the AFW flow to the affected SG by closing the AFW train discharge valves or stopping a pump, flow from the remaining AFW train or the SAFW System is directed to the intact SG for decay heat removal. This loss of condensate is partially compensated for by the retention of inventory in the intact SG.

(13) For cooldowns following loss of all onsite and offsite AC electrical power, the CSTs contain sufficient inventory to provide a minimum of 2 hours of decay heat removal as required by NUREG-0737 (Ref. 4), item II.E.1.1. This beyond DBA requirement provides more limiting criteria for CST inventory.

The CSTs satisfy Criterion 3 of the NRC Policy Statement.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for at least 10 minutes following a loss of MFW event from 102% RTP. After this time period, the accident analyses assume that AFW pump suction can be transferred to the safety related suction source (i.e., the SW System).

(continued)

BASES

LCO
(continued)

The required CST water volume is $\geq 22,500$ gallons, which is based on the need to provide at least 2 hours of decay heat removal following loss of all AC electrical power. The CSTs are considered OPERABLE when at least 22,500 gallons of water is available. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. Since the CSTs are 30,000 gallon tanks, only one CST is required to meet the minimum required water volume for this LCO.

The OPERABILITY of the CSTs is determined by maintaining the tank level at or above the minimum required water volume.

APPLICABILITY

In MODES 1, 2, and 3, the CSTs are required to be OPERABLE to support the AFW System requirements.

In MODE 4, 5, or 6, the CST is not required because the AFW System is not required to be OPERABLE.

ACTIONS

A.1 and A.2

If the CST water volume is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the preferred AFW pumps are OPERABLE and immediately available upon AFW initiation, and that the backup supply has the required volume of water available. Alternate sources of water include, but is not limited to, the SW System and the all-volatile-treatment condensate tank. In addition, the CSTs must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Continued verification of the backup supply is not required due to the large volume of water typically available from these alternate sources. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

(continued)



BASES

ACTIONS
(continued)

B.1 and B.2

If the backup supply cannot be verified or the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 22,500 gal minimum volume is met if one CST is ≥ 21 ft or if both CSTs are ≥ 12.5 ft. The 12 hour Frequency is based on operating experience and the need for operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. UFSAR, Section 10.7.4.
 2. UFSAR, Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
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B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the CCW System also provides this function for various safety related and nonsafety related components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water (SW) System, and thus to the environment. The safety related functions of the CCW system are covered by this LCO.

(101)
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The CCW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump and heat exchanger trains (Ref. 1) (see Figure B 3.7.7-1). Each CCW train consists of a manual suction and discharge valve, a pump, and a discharge check valve. The CCW loop header begins at the common piping at the discharge of the CCW trains and contains discharge to a common header which then supplies two parallel heat exchangers, either of which can supply the safety related and non-safety related components cooled by CCW. The CCW loop header begins at the common piping at the discharge of the two parallel heat exchangers, and continues up to the first isolation valve for each component supplied by the CCW System. The CCW loop header then continues from the last isolation valve on the discharge of each supplied load to the common piping at the suction of the CCW pumps. Each pump is powered from a separate Class 1E electrical bus. An open surge tank in the system provides for thermal expansion and contraction of the CCW system and ensures that sufficient net positive suction head is available to the pumps. The CCW System is also provided with a radiation detector (R-17) to isolate the surge tank from the Auxiliary Building environment and to provide indication of a leak of radioactive water into the CCW System.

The CCW System is normally maintained below 100°F by the use of one pump train in conjunction with one heat exchanger.

(continued)

BASES

The standby CCW pump will automatically start if the system pressure falls to 50 psig.

(continued)

BASES

(continued)

BASES

BACKGROUND
(continued)

The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Since the removal of decay heat via the RHR System is only performed during the recirculation phase of an accident, the CCW pumps do not receive an automatic start signal. Following the generation of a safety injection signal, the normally operating CCW pump will remain in service unless an undervoltage signal is present on either Class 1E electrical Bus 14 or Bus 16 at which time the pump is stripped from its respective bus. A CCW pump can then be manually placed into service prior to switching to recirculation operations which would not be required until a minimum of 46 minutes following an accident.

APPLICABLE
SAFETY ANALYSES

The design basis of the CCW System is for one CCW train and one CCW heat exchanger to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase. The Emergency Core Cooling System (ECCS) and containment models for a LOCA each consider the minimum performance of the CCW System. The normal temperature of the CCW is ≤ 100 °F, and, during LOCA conditions, a maximum temperature of 120°F is assumed. This prevents the CCW System from exceeding its design temperature limit of 200°F, and provides for a gradual reduction in the temperature of containment sump fluid as it is recirculated to the Reactor Coolant System (RCS) by the ECCS pumps. The CCW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

(169)

The CCW trains, heat exchangers, and loop headers are normally placed into service prior to the recirculation phase of an accident (i.e., 46 minutes following a large break event).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The CCW System can also function to cool the plant from RHR entry conditions ($T_{avg} < 350^{\circ}\text{F}$), to MODE 5 ($T_{avg} < 200^{\circ}\text{F}$), during normal cooldown operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR trains operating. Since CCW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in draining the CCW System within a short period of time. The CCW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of CCW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the Auxiliary Feedwater System) with acceptable results (Ref. 1). Leaks within the CCW System during post accident conditions can be mitigated by the available makeup water sources.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

LCO

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In the event of a DBA, one CCW train, one heat exchanger, and the loop header is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water (see Figure B 3.7.7-1). To ensure this requirement is met, two trains of CCW, two heat exchangers, and the loop header must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

101

A CCW train is considered OPERABLE when the pump is OPERABLE and capable of providing cooling water to the loop header. The automatic start logic associated with low CCW system pressure is not required for this LCO. In addition, if a CCW pump fails an Inservice Testing Program surveillance (e.g., pump developed head) the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses.

(continued)

BASES

(continued)

BASES

LCO
(continued)

189

The CCW loop header is considered OPERABLE when the associated piping, valves, ~~one of two CCW heat exchangers~~, surge tank, and the instrumentation and controls required to provide cooling water to the following safety related components are available and capable of performing their safety related function:

- a. Two RHR heat exchangers;
- b. Two RHR pump mechanical seal coolers and bearing water jackets;
- c. Three safety injection pump mechanical seal coolers; and
- d. Two containment spray pump mechanical seal coolers.

The CCW loop header temperature must also be $\leq 120^{\circ}\text{F}$ prior to the CCW cooling water reaching the first isolation valve supplying these components.

169

The CCW trains, ~~heat exchangers~~, and loop header are considered OPERABLE when they can be placed into service within the time limits assumed by the accident analyses (i.e., 46 minutes).

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The CCW loop header begins at the common piping at the discharge of the CCW ~~pump trains~~, through ~~one of two CCW heat exchangers~~, and ~~continues~~ up to the first isolation valve for each of the above components. The CCW loop header then continues from the last isolation valve on the discharge of each of the above components to the common piping at the suction of the CCW pumps.

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~~Only one of the two CCW heat exchangers is required since the heat exchanger is a passive device similar to the loop header piping.~~ The portion of CCW piping, valves, instrumentation and controls between the isolation valves to components a through d above is addressed by the following LCOs:

- a. LCO 3.4.6, "RCS Loops - MODE 4,"
- b. LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"
- c. LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"

(continued)

BASES

LCO
(continued)

- d. LCO 3.5.2, "ECCS - MODES 1, 2, and 3,"
- e. LCO 3.5.3, "ECCS - MODE 4,"
- f. LCO 3.9.3, "RHR and Coolant Circulation - Water Level \geq 23 Ft," and
- g. LCO 3.9.4, "RHR and Coolant Circulation - Water Level $<$ 23 Ft."

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The CCW piping inside containment for the reactor coolant pumps (RCPs) and the reactor support coolers also serves as a containment isolation barrier boundary. This is addressed by LCO 3.6.3, "Containment Isolation Barriers Boundaries."

The CCW system radiation detector (R-17) is not required to be OPERABLE for this LCO since the CCW system outside containment is not required to be a closed system.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be capable to perform its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling and containment integrity during the recirculation phase following a LOCA.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by LCO 3.4.7, LCO 3.4.8, LCO 3.9.3, and LCO 3.9.4.

(continued)

BASES

ACTIONS

A.1

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE CCW train could result in loss of CCW function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1-and-B

~~If one CCW heat exchanger is inoperable, action must be taken to restore OPERABLE status within 31 days. 2~~

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~~If the CCW train cannot be restored to OPERABLE status within the associated Completion Time in this Condition, the plant must be placed in a MODE in which the LCO does not apply. remaining OPERABLE heat exchanger is adequate to perform the heat removal function. To achieve this status however, the plant must be placed overall reliability is reduced because a passive failure in at least MODE 3 within 6 hours and the OPERABLE CCW heat exchanger could result in MODE 5 within 36 hours a loss of CCW function. The allowed 31 day Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner the redundant capabilities afforded by the OPERABLE train, and without challenging plant systems the low probability of a passive failure of the remaining heat exchanger.~~

C.1, and C.2

~~If the CCW train or CCW heat exchanger cannot be restored to OPERABLE status within the associated Completion Time, and the plant must be placed in a MODE in which the LCO does not apply. 3~~

~~With both CCW trains or the loop header inoperable, action must be immediately initiated to restore OPERABLE. To achieve~~

(continued)

BASES

189

~~This status to one CCW train or the loop header, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

(continued)



BASES

ACTIONS D.1, D.2, and D.3
(continued):

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With both CCW trains, both CCW heat exchangers, or the loop header inoperable, action must be immediately initiated to restore OPERABLE status to one CCW train, one CCW heat exchanger, and the loop header. In this Condition, there is no OPERABLE CCW System available to provide necessary cooling water which is a loss of a safety function. Also, the plant must be placed in a MODE in which the consequences of a loss of CCW coincident with an accident are reduced. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The plant is not required to exit the Applicability for this LCO (i.e., enter MODE 5) until at least one CCW train or, one CCW heat exchanger, and the loop header is restored to OPERABLE status to support RHR operation.

REQUIREMENTS

required by other LCOs

100

~~SURVEILLANCE~~ ~~SR 3.7.7.1~~

~~Verifying the correct alignment for manual and power operated valves in the CCW flow path servicing post accident related equipment provides assurance that the proper flow paths exist for CCW operation~~ Required Actions D.1, D.2, and D.3 are modified by a Note indicating that all required MODE changes or power reductions are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

(continued)

BASES (continued)

**SURVEILLANCE
REQUIREMENTS**

SR 3.7.7.1

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Verifying the correct alignment for manual and power operated valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW loop header.

SR 3.7.7.2

This SR verifies that the two motor operated isolation valves to the RHR heat exchangers (738A and 738B) can be operated when required since the valves are normally maintained closed. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 2).

REFERENCES

1. UFSAR, Section 9.2.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW system also provides this function for various safety related and nonsafety related components. The safety related functions of the SW System are covered by this LCO.

The SW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump trains (Ref. 1) (see Figure B 3.7.8-1). The physical design of the SW System is such that one 100% capacity pump from each class 1E electrical bus (Buses 17 and 18) is arranged on a common piping header which then supplies the SW loop header. For the purposes of this LCO, a SW train is based on electrical source only.

Each train is powered from a separate Class 1E electrical bus and consists of two 100% capacity pumps and associated discharge check valves and manual isolation valves. The SW loop header begins from the discharge of the trains and supplies the safety related and nonsafety related components cooled by SW. The pumps in the system are normally manually aligned. One pump in each train is selected to automatically start upon receipt of an undervoltage signal on its respective bus. Upon receipt of a safety injection signal, each SW pump will automatically start in a predetermined sequence.

The SW loop header supplies the cooling water to all safety related and nonsafety related components. The nonsafety related and long-term safety functions (e.g., component cooling water heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. These valves automatically close on a coincident safety injection signal and undervoltage signal on Buses 14 and 16.

(continued)

BASES

BACKGROUND
(continued)

The suction source for the SW System is the screenhouse which is a seismic structure located on Lake Ontario. The discharge from the SW System supplied loads returns back to Lake Ontario. The principal safety related functions of the SW system is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System, provide cooling water to the diesel generators (DGs) and containment recirculation fan coolers (CRFCs) and to provide a safety related source of water to the Auxiliary Feedwater (AFW) System.

APPLICABLE
SAFETY ANALYSES

The design basis of the SW System is for one SW train in conjunction with a 100% capacity containment cooling system (i.e., CRFC) to provide for heat removal following a steam line break (SLB) inside containment to ensure containment integrity. The SW System is also designed, in conjunction with the CCW System and a 100% capacity Emergency Core Cooling System and containment cooling system, to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is recirculated to the Reactor Coolant System by the ECCS pumps. The SW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

Following the receipt of a safety injection signal, all four SW pumps are designed to start (if not already running) to supply the system loads. If a coincident safety injection and undervoltage signal occurs, then each nonsafety related and nonessential load within the SW System is isolated by redundant motor operated valves that are powered by separate Class 1E electrical trains. The SW pumps are sequenced to start within 17 seconds following a safety injection signal. The selected SW pumps are sequenced to start after a 40 second time delay following an undervoltage signal energization of the electrical bus supplying the selected pump (i.e., Bus 17 or Bus 18) after an undervoltage signal.

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(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The SW system, in conjunction with the CCW System, can also cool the plant from residual heat removal (RHR) entry conditions ($T_{avg} < 350^{\circ}\text{F}$) to MODE 5 ($T_{avg} < 200^{\circ}\text{F}$) during normal operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR System trains that are operating. Since SW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in failing the SW System to potentially multiple safety related functions. The SW system has been evaluated to demonstrate the capability to meet cooling needs with an assumed 500 gal leak. The SW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of SW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the AFW Systems) with acceptable results (Ref. 1).

The temperature of the fluid supplied by the SW System is also a consideration in the accident analyses. If the cooling water supply to the containment recirculation fan coolers and CCW heat exchangers is too warm, the accident analyses with respect to containment pressure response following a SLB and the containment sump fluid temperature following a LOCA may no longer be bounding. If the cooling water supply is too cold, the containment heat removal systems may be more efficient than assumed in the accident analysis. This causes the backpressure in containment to be reduced which potentially results in increased peak clad temperatures.

The SW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one SW train and the loop header is required to be OPERABLE to provide the minimum heat removal capability to ensure that the system functions to remove post accident heat loads as assumed in the safety analyses. To ensure this requirement is met, two trains of SW and the loop header must be OPERABLE (see Figure B 3.7.8-1). At least one SW train will operate assuming that the worst case single active failure occurs coincident with the loss of offsite power.

(continued)

BASES

LCO
(continued)

A SW train is defined based on electrical power source such that SW Pumps A and C form one train and SW Pumps B and D form the second train. A SW train is considered OPERABLE when one pump in the train is OPERABLE and capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW pumps:

- a. Level \geq 5 feet; and
- b. Temperature \geq 35°F above 50% RTP and \leq 80°F.

The lower screenhouse bay temperature is only specified above 50% RTP since this value is only a consideration when evaluating LOCA at or near full power conditions. In addition, if a SW pump fails on Inservice Testing Program surveillance (e.g., pump developed head), the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses (Ref. 1).

An OPERABLE SW train also requires that all nonessential and nonsafety related loads can be isolated by the six motor operated isolation valves which are powered from the same Class 1E electrical train as the pumps. Therefore, motor operated valves 4609, 4614, 4615, 4616, 4663, and 4670 must be OPERABLE and capable of closing for SW Pumps A and C while valves 4613, 4664, 4733, 4734, 4735, and 4780 must be OPERABLE and capable of closing for SW Pumps B and D.

The SW loop header is considered OPERABLE when the associated piping, valves, and the instrumentation and controls required to provide cooling water from each OPERABLE SW train to the following safety related components are available and capable of performing their safety related function:

- a. Four CRFCs;
- b. Two CCW heat exchangers;

(continued)

BASES

LCO
(continued)

- c. Two DGs;
- d. Three preferred AFW pumps;
- e. Two standby AFW pumps; and
- f. Three safety injection pump bearing housing coolers.

An OPERABLE SW loop header also requires a flowpath flow path through the diesel generator (4665, 4760, 4669, and 4668B) and CRFC (4756, 4623, 4640, 4756 and 4639) cross-ties.

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The SW trains and loop header are ~~considered OPERABLE when they can supply:~~

assumed to supply the following components following an accident!

- a. The CRFCs, DGs and safety injection pump bearing housing coolers immediately following a safety injection signal (i.e., after the loop header becomes refilled);
- b. The preferred AFW and SAFW pumps within 10 minutes following receipt of a low SG level signal; and
- c. The CCW heat exchangers within 46 minutes following a safety injection signal.

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The SW loop header begins at the common piping at the discharge of both SW pump trains and ends at the first isolation valve for each of the above components. Since the SW System discharges back to Lake Ontario, the cooling water flowpath flow path through the above components and subsequent discharge is addressed under their respective LCO. This includes:

- a. LCO 3.5.2, "ECCS - MODES 1, 2, and 3;"
- b. LCO 3.5.3, "ECCS - MODE 4;"
- c. LCO 3.6.6, "CS, CRFC, and Post-Accident Charcoal Systems;"
- d. LCO 3.7.5, "AFW Systems;"
- e. LCO 3.7.7, "CCW System;"

(continued)

BASES

LCO
(continued)

- f. LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4;" and
- g. LCO 3.8.2, "AC Sources - MODES 5 and 6."

12a

The SW piping inside containment for the CRFCs and the reactor compartment coolers also serves as a containment isolation barrier boundaries. This is addressed under LCO 3.6.3, "Containment Isolation Barriers Boundaries."

APPLICABILITY

In MODES 1, 2, 3, and 4, the SW System is a normally operating system which must be capable of performing its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling during the recirculation phase following a LOCA or loss of containment integrity following a SLB.

In MODES 5 and 6, the OPERABILITY requirements of the SW system are determined by LCO 3.6.6, LCO 3.7.7, and LCO 3.8.2.

ACTIONS

A.1

If one SW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW train could result in loss of SW System function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

(continued)

BASES

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If the SW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, G

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~~With both SW trains or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.~~

~~2, and Required Action C.3~~

~~With both SW trains or the loop header inoperable, action must be modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be immediately initiated to restore OPERABLE status to one SW train or the loop header entered for the component cooling water heat exchanger made inoperable by SW. In this Condition, there is no OPERABLE. This note is provided since the inoperable SW System available to provide necessary cooling water which is the loss of a safety functions system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.—~~

~~Also, the plant must be placed in a MODE in which the consequences of a loss of SW coincident with~~

~~an~~

SURVEILLANCE SR 3.7.8.1
REQUIREMENTS

142

~~This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses. To achieve this status, the plant must be~~

(continued)

BASES

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~~placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The 24 hour frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.~~—

(continued)

BASES

~~The allowed Completion Times are reasonable, based on operating experience~~

~~SURVEILLANCE SR 3.7.8.2~~
REQUIREMENTS

(continued)

135

~~Verifying the correct alignment for manual, to reach the required plant conditions from full power conditions in an orderly manner operated, and without challenging plant systems automatic valves in the SW flow path provides assurance that the proper flow paths exist for SW operation. The plant should not exit the Applicability for this LCO (i.e., enter MODE 5) until at least one SW train or the loop header is restored to OPERABLE status to support RHR operation.~~

(continued)



BASES

SURVEILLANCE — SR 3.7.8.1
REQUIREMENTS

135

~~Verifying the correct alignment for manual, power operated, and automatic valves in the SW flow path servicing post-accident related equipment provides assurance that the proper flow paths exist for SW operation. This includes verification of the SW cross connect valves. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.~~

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the SW flow to individual components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

REQUIREMENTS
(continued)

SURVEILLANCE — SR 3.7.8.2

3.7.8.3

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~~This SR verifies proper automatic operation of the that all SW motor operated isolation loop header cross-tie valves on an actual or simulated actuation signal are locked in the correct position. This includes verification that manual valves 4623, 4639, 4640, 4665, 4668B, 4669, 4756, and 4760 are locked open and that manual valves 4610, 4611, 4612, and 4779 are locked closed. The 31 day Frequency is based on engineering judgement, is consistent with the procedural controls governing locked valves, and ensures correct valve positions.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS SR 3.7.8.4

(continued)

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This SR verifies proper automatic operation of the SW motor operated isolation valves on an actual or simulated actuation signal (i.e., coincident safety injection and undervoltage signal). SW is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3-3.7.8.5

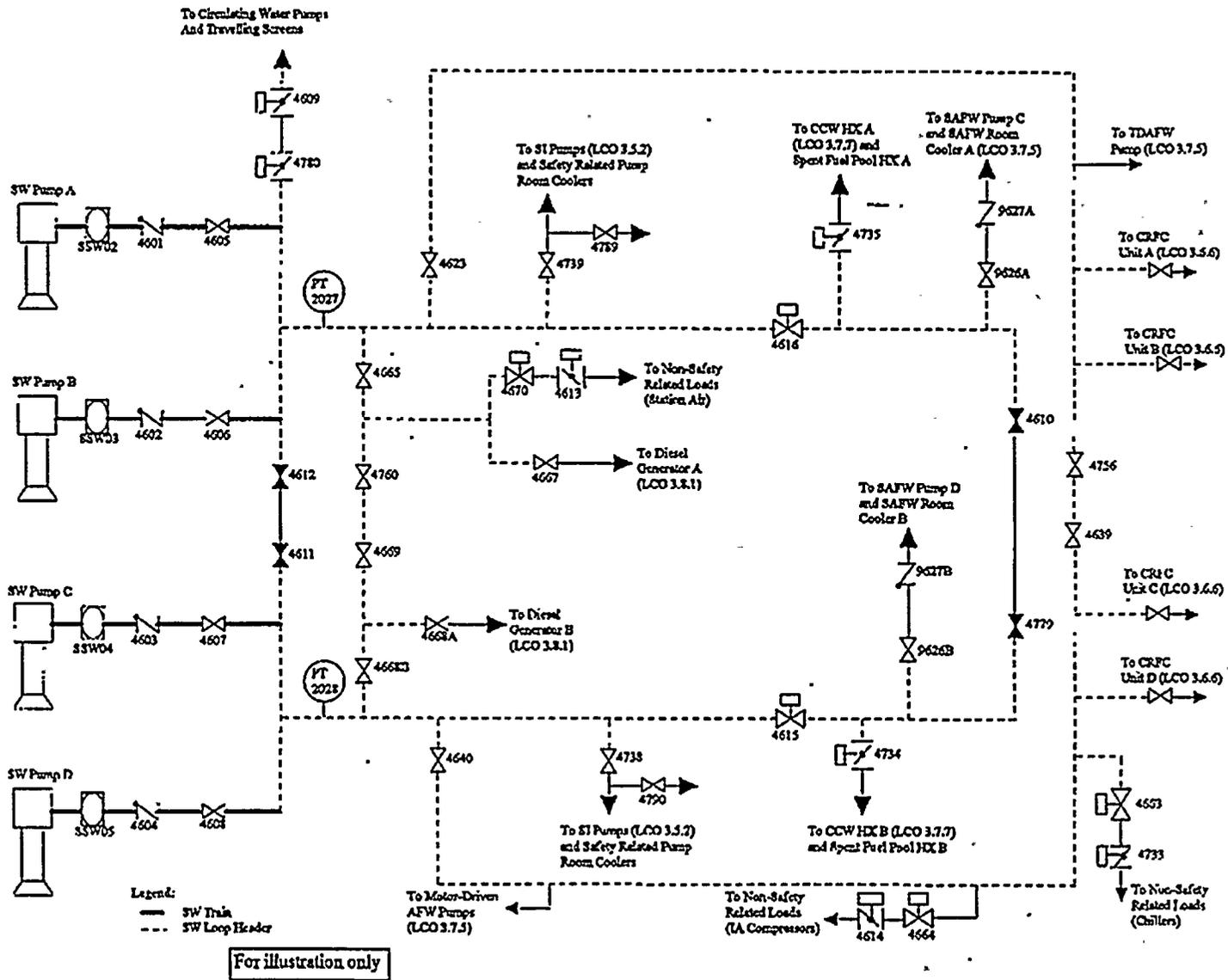
169

This SR verifies proper automatic operation of the SW pumps on an actual or simulated actuation signal. This includes the actuation of the SW pumps following an undervoltage signal and following a coincident safety injection and undervoltage signal. SW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.1.
 2. UFSAR, Section 6.2.
-

Figure B 3.7.8-1
SW System



B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Air Treatment System (CREATS)

BASES

BACKGROUND

According to Atomic Industry Forum (AIF) GDC 11 (Ref. 1), a control room shall be provided which permits continuous occupancy under any credible postaccident condition without excessive radiation exposures of personnel. Exposure limits are provided in GDC 19 of 10 CFR 50, Appendix A (Ref. 2) which requires that control room personnel be restricted to 5 rem whole body, or its equivalency, for the duration of the accident. The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity for 30 days without exceeding this 5 rem whole body limit. The CREATS is part of the Control Building ventilation system.

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The CREATS consists of a high efficiency particulate air (HEPA) filter, activated charcoal ~~absorbers~~ adsorbers for removal of gaseous activity (principally iodines), and two fans (control room return air fan and emergency return air fan) (see Figure B 3.7.9-1). Ductwork, dampers, and instrumentation also form part of the system as well as demisters to remove water droplets from the air stream (Ref. 3).

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The CREATS is an emergency system, parts of which may operate during normal plant operations. Actuation of the CREATS places the system in one of five separate states of the emergency mode of operation, depending on the initiation signal. The following are the normal and emergency modes of operation for the CREATS:

CREATS Mode A

The CREATS is in the standby mode with the exception that the control room return air fan is in operation.

(continued)



BASES

BACKGROUND
(continued)

CREATS Mode B

This is the CREATS configuration following an accident with a radiation release as detected by radiation monitor R-1. Upon receipt of an actuation signal, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm (approximately one fourth of the Control Building Ventilation System design) through the CREATS charcoal and HEPA filters. All outside air that enters the CREATS, as controlled by an air adjust switch (S-81), is also circulated through the CREATS charcoal and HEPA filters.

CREATS Mode C

This is the same CREATS configuration as Mode B with the exception that all outside air is isolated to the control room by one damper in each air supply flow path.

CREATS Mode D

This is the CREATS configuration following the detection of smoke within the Control Building. Upon receipt of an actuation signal, the system continues to draw outside air. However, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm through the CREATS and HEPA filters. This effectively purges the control room air environment.

CREATS Mode E

This is the same CREATS configuration as Mode D with exception that all outside air is isolated to the control room by one damper in each air supply flow path.

(continued)

BASES

BACKGROUND
(continued)CREATS Mode F

This is the CREATS configuration following the detection of a toxic gas as indicated by the chlorine or ammonia detectors, or high radiation as detected by R-36 (gas), R-37 (particulate), or R-38 (iodine). Upon receipt of an actuation signal, the system aligns itself consistent with Mode C except that two dampers in each air supply path are isolated.

Normally open air supply isolation dampers are arranged in series so that the failure of one damper to close will not result in a breach of isolation.

The air entering the control room is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. The actions of the toxic gas and high radiation state (Mode F) are more restrictive, and will override the actions of the emergency radiation state (Mode B or C). Only the high radiation state CREATS Mode F is addressed by this LCO.

APPLICABLE
SAFETY ANALYSES

The location of components and CREATS related ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 3). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

In MODES 5 and 6, and during movement of irradiated fuel assemblies, the CREATS ensures control room habitability in the event of a fuel handling accident or waste gas decay tank rupture accident.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The CREATS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The CREATS is comprised of a filtration train and two independent and redundant isolation damper trains all of which are required to be OPERABLE. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

The CREATS is considered OPERABLE when the individual components necessary to permit CREATS Mode F operation are OPERABLE (see Figure B 3.7.9-1). The CREATS filtration train is OPERABLE when the associated:

- a. Control room return air and emergency return air fans are OPERABLE and capable of providing forced flow;
- b. HEPA filters and charcoal ~~absorbers~~ ~~adsorbers~~ for the emergency return air fan are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. Ductwork, valves, and dampers (including AKD06 and AKD09) are OPERABLE, and air circulation can be maintained.

The CREATS isolation dampers are considered OPERABLE when the damper (AKD01, AKD04, AKD05, AKD08, and AKD10) can close on an actuation signal to isolate outside air or is closed with motive force removed. Two dampers are provided for each outside air path.

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(continued)

BASES

LCO
(continued)

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors. Opening of the access doors for entry and exit does not violate the control room boundary. An access door may be opened for extended periods provided a dedicated individual is stationed at the access door to ensure closure, if required (i.e., the individual performs the isolation function), the door is able to be closed within 30 seconds upon indication of the need to close the door, and the CREATS filtration train is OPERABLE.

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APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS must be OPERABLE to control operator exposure during and following a DBA.

In MODE 5 or 6, the CREATS is required to cope with the release from the rupture of a waste gas decay tank.

During movement of irradiated fuel assemblies, the CREATS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1 and A.2

With the CREATS filtration train inoperable, action must be taken to restore OPERABLE status within 48 hours or isolate the control room from outside air. In this Condition, the isolation dampers are adequate to perform the control room protection function but no means exist to filter the release of radioactive gas within the control room. The 48 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to isolate the control room.

Required Action A.2 is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

(continued)

BASES

ACTIONS
(continued)B.1

With one CREATS isolation damper inoperable for one or more outside air flowpaths, flow paths, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREATS isolation damper is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREATS isolation damper could result in loss of CREATS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining isolation damper to provide the required isolation capability.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1 , D.22.1, and D.32.2

In MODE 5 or 6 or during movement of irradiated fuel assemblies, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, action must be taken to immediately place the OPERABLE isolation damper(s) in CREATS Mode F. This action ensures that the remaining damper(s) are OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

(continued)

BASES

~~In addition, action must be immediately taken to suspend activities that could result in a release of radioactivity that might enter the control room.~~ ACTIONS D.1, D.2.1, and D.2.2 (continued)

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An alternative to Required Action D.1 is immediately suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel or other components to a safe position.

ACTIONS—E.1

—(continued)

In MODE 1, 2, 3, or 4, if both CREATS isolation dampers for one or more outside air supply flow paths are inoperable, the CREATS may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Failure of the integrity of the control room boundary (i.e., walls, floors, ceilings, ductwork or access doors) also results in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

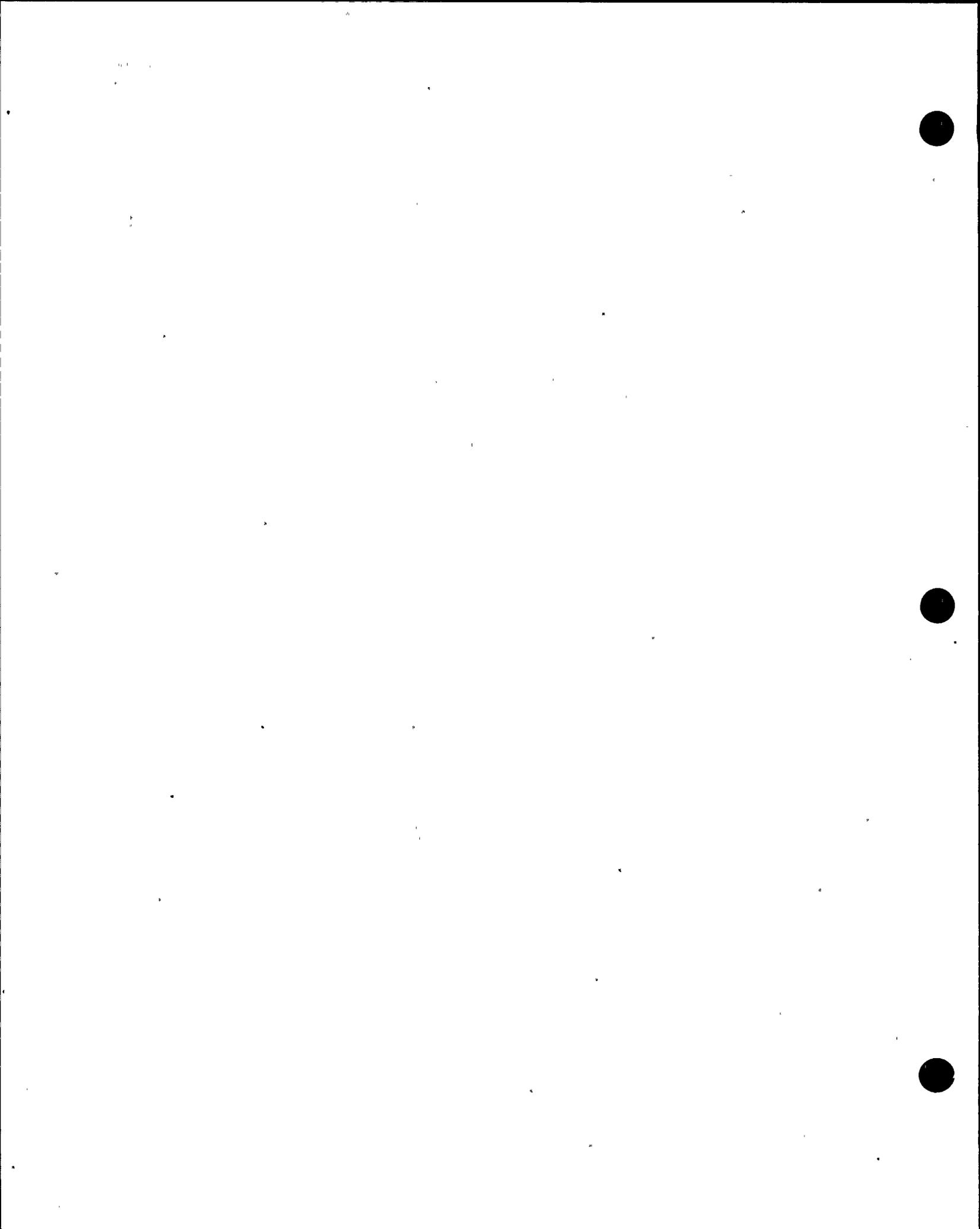
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F.1, and F.2, and F.3

In MODE 5 or 6, or during movement of irradiated fuel assemblies with two CREATS isolation dampers for one or more air supply outside air flow paths inoperable, action must be taken immediately to restore one isolation damper in each affected air supply path to OPERABLE status. In addition, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes accident risk. This does not preclude the movement of fuel or other components to a safe position.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each CREATS filtration train once every 31 days for ≥ 15 minutes provides an adequate check of this system. The 31 day Frequency is based on the reliability of the equipment.

~~SURVEILLANCE~~ SR 3.7.9.2

REQUIREMENTS

(continued)

This SR verifies that the required CREATS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREATS filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing the performance of the HEPA filter, charcoal ~~absorber~~ adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The minimum required flowrate through the CREATS filtration train is 2000 cubic feet per minute ($\pm 10\%$). Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 4).

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SR 3.7.9.3

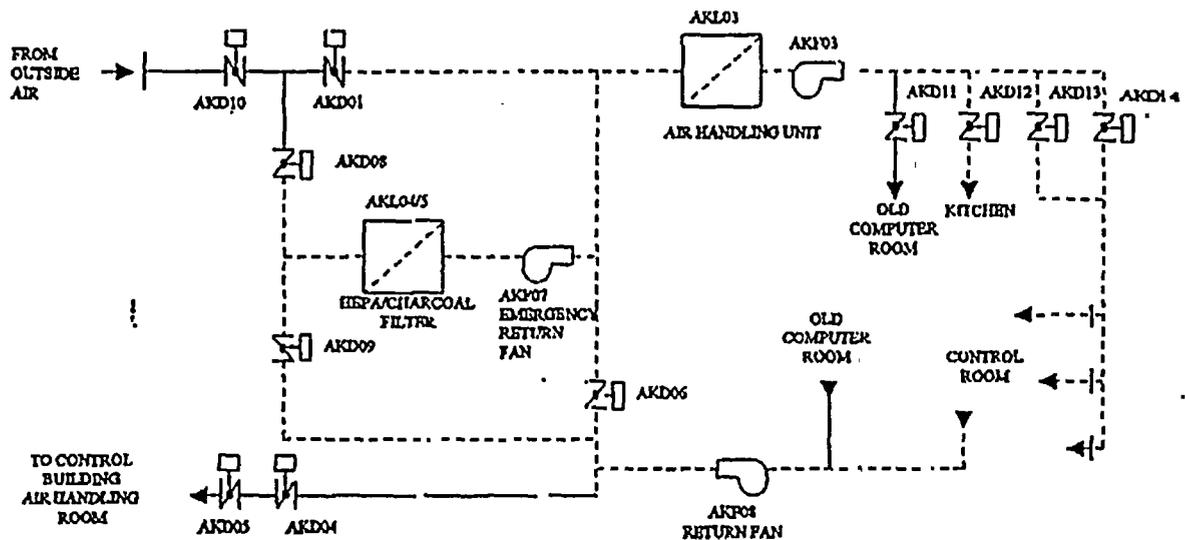
This SR verifies that the CREATS filtration train starts and operates and each CREATS isolation damper actuates on an actual or simulated actuation signal. The Frequency of 24 months is based on Regulatory Guide 1.52 (Ref. 4).

(continued)

BASES

REFERENCES

1. Atomic Industry Forum (AIF) GDC 11, Issued for comment July 10, 1967.
 2. 10 CFR 50, Appendix A, GDC 19.
 3. UFSAR, Section 6.4.
 4. Regulatory Guide 1.52, Revision 2.
-



Legend:

---- CREATS Filtration Train

For illustration only

Notes:

1. Outside air flowpath isolation dampers includes AKD01, AKD04, AKD05, AKD08, and AKD10.
2. The CREATS filtration train does not include the air handling unit (AKL03 and AKF03).

Figure B 3.7.9-1
CREATS

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B 3.7 PLANT SYSTEMS

B 3.7.10 Auxiliary Building Ventilation System (ABVS)

BASES

BACKGROUND

The ABVS filters airborne radioactive particulates from the area of the spent fuel pool (SFP) following a fuel handling accident. The ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the Auxiliary Building including the SFP area.

The ABVS consists of an air handling unit, a series of exhaust fans, charcoal filters, ductwork, and dampers (Ref. 1). The exhaust fans include the following fans which all discharge into a common ductwork that supplies the Auxiliary Building main exhaust fans A and B (see Figure B 3.7.10-1):

- 106
- a. Intermediate Building exhaust fans A and B;
 - b. Auxiliary Building exhaust fan C;
 - c. Auxiliary Building charcoal filter fans A and B;
 - d. Auxiliary Building exhaust fan G; and
 - e. Control access exhaust fans A and B.

The only components which filter the environment associated with the SFP are the Auxiliary Building main exhaust fans and Auxiliary Building exhaust fan C. Therefore, these are the only fans considered with respect to the ABVS in this LCO.

(continued)

BASES (continued)

BACKGROUND
(continued)

93

Auxiliary Building exhaust fan C takes suction from the SFP and decontamination pit areas on the operating level of the Auxiliary Building. The air is first drawn through the SFP Charcoal Adsorber System which consists of roughing filters and charcoal ~~absorbersadsorbers~~. The roughing filters protect the charcoal ~~absorbersadsorbers~~ from being fouled with dirt particles while the charcoal ~~absorbersadsorbers~~ remove the radioactive iodines from the atmosphere. Auxiliary Building exhaust fan C then discharges into the common ductwork that supplies the Auxiliary Building main exhaust fans. This common ductwork contains a high efficiency particulate air (HEPA) filter which is not credited in the dose analyses.

The Auxiliary Building main exhaust fans are each 100% capacity fans which can maintain a negative pressure on the operating floor of the Auxiliary Building through orientation of the system dampers. This negative pressure causes air flow on the operating floor to be toward the SFP which ensures that air in the vicinity of the SFP is first filtered through the SFP Charcoal Adsorber System. The Auxiliary Building main exhaust fans and exhaust fan C are powered from non-Engineered Safeguards Features buses.

The Auxiliary Building main exhaust fans discharge to the plant vent stack. The plant vent stack is continuously monitored for noble gases (R-14), particulates (R-13) and iodine (R-10B). During normal power operation, the ABVS is placed in the "out" mode by the interlock mode switch where "out" defines the status of the SFP charcoal filters. This causes all exhaust fans without any HEPA or charcoal filters (excluding the Auxiliary Building Main exhaust fans) and Auxiliary Building exhaust fan C to trip upon a signal from R-10B, R-13 or R-14 to stop the release of any radioactive gases. During fuel movement within the Auxiliary Building, the interlock mode switch is placed in the "in" mode such that only exhaust fans without any HEPA or charcoal filters (excluding Auxiliary Building main exhaust fans) are tripped.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

93

The ABVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that Auxiliary Building exhaust fan C, the SFP Charcoal Adsorber System, and one Auxiliary Building main exhaust fan ~~is functional~~ are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the minimum filtration system components which result in offsite doses well within the limits of 10 CFR 100 (Ref. 3). The failure of any or all of these filtration system components results in doses which are slightly higher but still within 10 CFR 100 limits. The fuel handling accident assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

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The remainder of the ABVS described in the Background is not required for any DBA since it is non-safety related and supplied only from offsite power sources.

The ABVS satisfies Criterion 3 of the NRC Policy Statement.

LCO

10b

The ABVS is required to be OPERABLE to ensure that offsite doses are well within the limits of 10 CFR 100 (Ref. 3) following a fuel handling accident in the Auxiliary Building. The failure of the ABVS coincident with a fuel handling accident results in doses which are slightly higher but still within 10 CFR 100 limits.

The ABVS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary Building following a fuel handling accident are OPERABLE and in operation (see Figure B 3.7.10-1). The ABVS is considered OPERABLE when its associated:

- a. Auxiliary Building exhaust fan C and either Auxiliary Building main exhaust fan A or B is OPERABLE and in operation;

(continued)

BASES (continued)

- ~~LCO~~ b. Auxiliary Building main exhaust fan HEPA filter and—
~~(continued)~~ SFP charcoal adsorbers are not excessively restricting flow, and the SFP Charcoal Adsorber System is capable of performing its filtration function;
- ~~LCO~~ c. Ductwork, valves, and dampers are OPERABLE, and air
~~(continued)~~ circulation and negative pressure can be maintained on the Auxiliary Building operating floor; and
- d. Interlock mode switch is placed in the "in" mode.
-

APPLICABILITY

During movement of irradiated fuel in the Auxiliary Building, the ABVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. The ABVS is only required when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. Any fuel handling accident which occurs after 60 days results in offsite doses which are well within 10 CFR 100 limits (Ref. 3) due to the decay rate of iodine.

Since a fuel handling accident can only occur as a result of fuel movement, the ABVS is not MODE dependant and only required when irradiated fuel is being moved.

ACTIONS

A.1

When the ABVS is inoperable, action must be taken to place the plant in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

(continued)

BASES (continued)

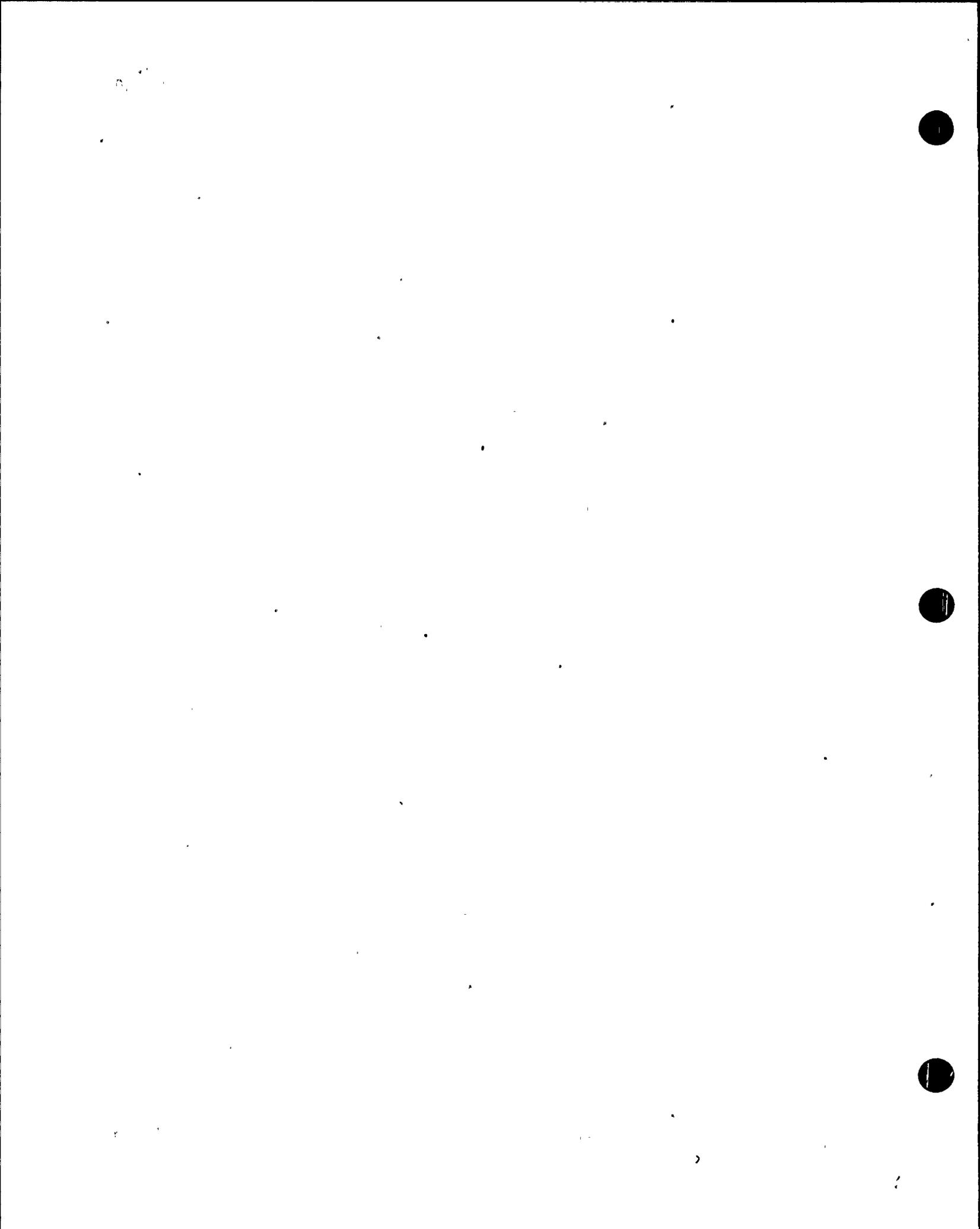
REQUIREMENTS

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~~SURVEILLANCE~~ — SR 3.7.10.1

This SR verifies the OPERABILITY of the ABVS and the integrity of the Auxiliary Building enclosure. ~~The ability of the Auxiliary Building to maintain negative pressure with respect to the uncontaminated outside environment is periodically verified to ensure proper function of the ABVS. This SR ensures that LCO 3.0.3 is not applicable. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure if moving irradiated fuel assemblies in the Auxiliary Building, to prevent unfiltered LEAKAGE which have decayed < 60 days since being irradiated, the fuel movement is independent of reactor operations. This SR ensures that Auxiliary Building exhaust fan C, and either Auxiliary Building main exhaust fan A or B are in operation. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a negative pressure is being maintained in the Auxiliary Building, and that the ABVS interlock mode switch is in the correct position reactor shutdown.~~ —

(continued)



BASES (continued)

The Frequency SURVEILLANCE SR 3.7.10.1
REQUIREMENTS

144

This SR verifies the OPERABILITY of 24 hours is based on engineering judgement and shown to be acceptable through operating experience the ABVS.

SR 3.7.10.2

144

This SR verifies that During fuel movement operations, the required SFP Charcoal Adsorber System testing ABVS is performed designed to maintain a slight negative pressure in accordance with the Ventilation Filter Testing Program (VFTP) the Auxiliary Building to prevent unfiltered LEAKAGE. The SFP Charcoal Absorber System filter tests This SR ensures that Auxiliary Building exhaust fan C, and either Auxiliary Building main exhaust fan A or B are in general accordance with Regulatory Guide 1.52 (Refoperation and that the ABVS interlock mode switch is in the correct position. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

SR 3.7.10.2

144

This SR verifies the integrity of the Auxiliary Building enclosure. The ability of the Auxiliary Building to maintain negative pressure with respect to the uncontaminated outside environment must be periodically verified to ensure proper functioning of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered leakage. This SR ensures that a negative pressure is being maintained in the Auxiliary Building. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS SR 3.7.10.3

(continued)

This SR verifies that the required SFP Charcoal Adsorber System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SFP Charcoal Adsorber System filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). There is no minimum required flowrate through the SFP charcoal adsorbers since SR 3.7.10.2 requires verification that a negative pressure is maintained during fuel movement in the Auxiliary Building. As long as this minimum pressure is maintained by drawing air from the surface of the SFP through the SFP charcoal adsorbers, the assumptions of the accident analyses are met. Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 5).

Q3

150

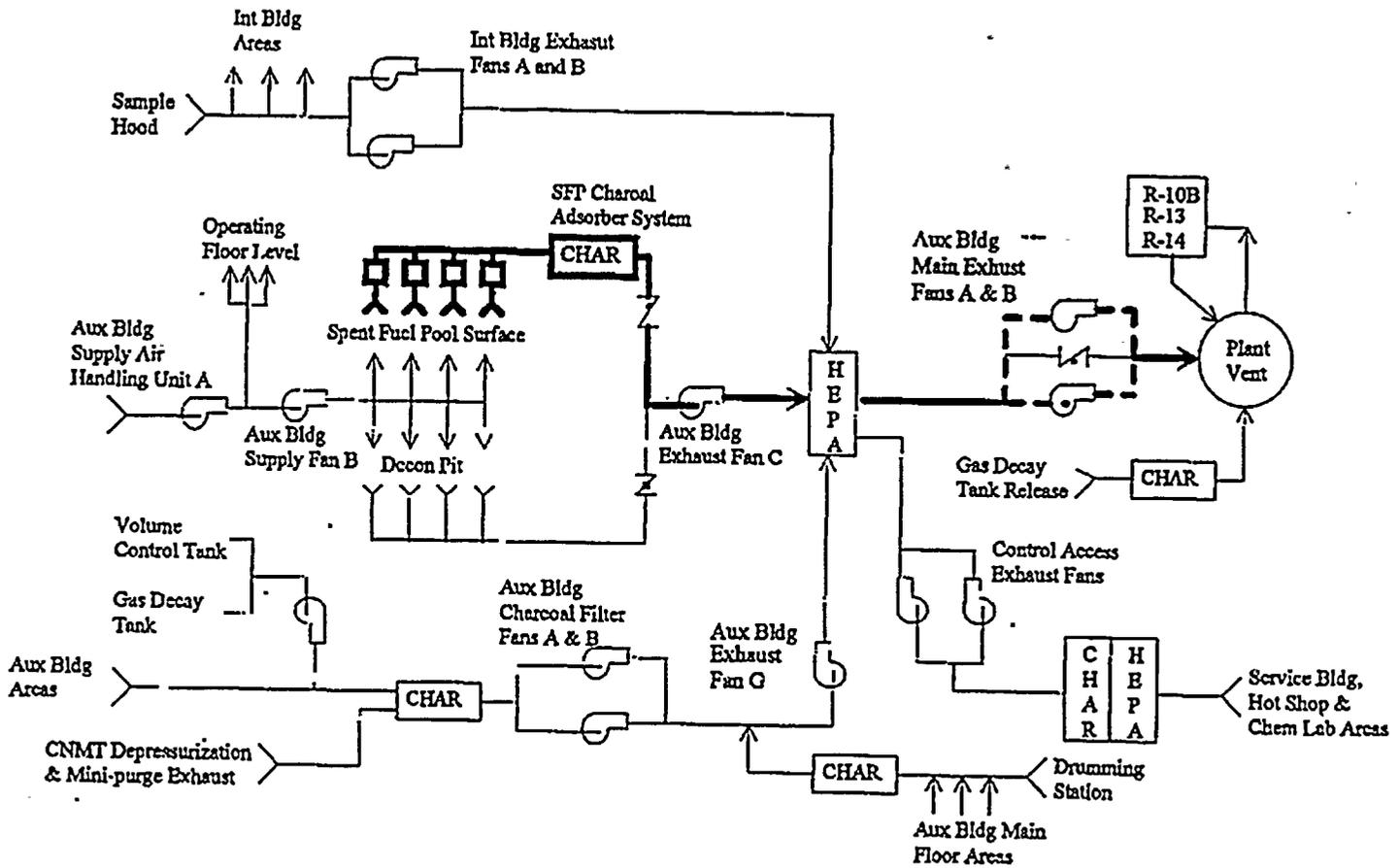
149

REFERENCES

1. UFSAR, Section 9.4.2.
 2. UFSAR, Section 15.7.3.2.
 3. 10 CFR 100.
 4. Regulatory Guide 1.25, Rev. 0.
 5. Regulatory Guide 1.52, Rev. 2.
-

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Figure B 3.7.10-1
ABVS



Legend:

- Flowpath required by LCO (Aux Bldg Exhaust Fan C HEPA filter not required for LCO but Aux Bldg operating floor must be at a negative pressure)
- - - 1 of 2 flowpaths required by LCO
- SFP Roughing filters

For illustration only



B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool (SFP) meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level provides protection against exceeding the offsite dose limits.

The SFP is a seismically designed structure located in the Auxiliary Building (Ref. 1). The pool is internally clad with stainless steel that has a leak chase system at each weld seam to minimize accidental drainage through the liner. The SFP is also provided with a barrier between the spent fuel storage racks and the fuel transfer system winch. This barrier, up to the height of the spent fuel racks, prevents inadvertent drainage of the SFP via the fuel transfer tube.

The SFP Cooling System is designed to maintain the pool $\leq 120^{\circ}\text{F}$ during normal conditions and refueling operations (Ref. 2). The cooling system normally takes suction near the surface of the SFP such that a failure of any pipe in the system will not drain the pool. The cooling system return line to the pool also contains a 0.25 inch vent hole located near the SFP surface level to prevent siphoning. Finally, control board alarms exist with respect to the SFP level and temperature. These features all help to prevent inadvertent draining of the SFP.

APPLICABLE
SAFETY ANALYSES

The minimum water level in the SFP is an assumption of the fuel handling accident described in the UFSAR (Ref. 3) and Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary as based on this assumption is a small fraction of the 10 CFR 100 (Ref. 5) limits.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

Based on the requirements of Reference 4, there must be 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water available, the assumptions of Reference 4 can be used directly. These assumptions include the use of a decontamination factor of 100 in the analysis for iodine. A decontamination factor of 100 enables the analysis to assume that 99% of the total iodine released from the pellet to cladding gap of all dropped fuel assembly rods is retained by the SFP water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.

In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel storage racks, however, there may be < 23 ft of water between the top of the fuel bundle and the surface, indicated by the width of the bundle and difference between the top of the rack and active fuel. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The SFP water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

The SFP water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required during movement of irradiated fuel assemblies within the SFP.

(continued)

BASES (continued)

APPLICABILITY This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists. Since a fuel handling accident can only occur during movement of fuel, this LCO is not applicable during other conditions. During refueling operations in MODE 6, the SFP water level (and boron concentration) are in equilibrium with the refueling water cavity. The water level under these conditions is then controlled by LCO 3.9.5, "Refueling Cavity Water Level" which requires the refueling cavity water level to be maintained ≥ 23 feet above the top of the reactor vessel flange. A refueling cavity water level of ≥ 23 feet above the top of the reactor vessel flange will result in > 23 feet of water above the top of the active fuel in the storage racks assuming that atmospheric pressure within containment and the Auxiliary Building are equivalent.

ACTIONS

A.1

When the initial conditions assumed in the fuel handling accident analysis cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.11.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically during movement of irradiated fuel assemblies to ensure the fuel handling accident assumptions are met. The ~~317~~ day Frequency is appropriate because the volume in the pool is normally stable and the SFP is designed to prevent drainage below 23 ft. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

Verification of SFP water level can be accomplished by several means. The top of the upper SFP pump suction line is 23 ft above the fuel stored in the pool. If there is \geq 23 ft of water above the reactor vessel flange (as required by LCO 3.9.5), with equal pressure in the containment and the Auxiliary Building, then at least 23 ft of water is available above the top of the active fuel in the storage racks.

In addition to the physical design features, there are two SFP level alarms (LAL 634) which are available to alert the operators of changing SFP level. A low level alarm will actuate when the SFP water level falls 4 inches or more from the normal level while a high level alarm will actuate when the SFP water level rises 4 inches or more from the normal level. These alarms must receive a calibration consistent with industry practices before they are to be used to meet this SR.

REFERENCES

1. UFSAR, Section 9.1.2.
 2. UFSAR, Section 9.1.3.
 3. UFSAR, Section 15.7.3.
 4. Regulatory Guide 1.25, Rev. 0.
 5. 10 CFR 100.11.
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B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool (SFP) normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both SFP regions is based on the use of unborated water such that configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded.

The double contingency principle discussed in ANSI N-16.1-1975 (Ref. 1) and Reference 2 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.173-7.13, "Fuel Assemblies ~~Spent Fuel Pool~~ (SFP) Storage." Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(169)

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Ref. 3 and 4). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by voiding (which would result in the addition of negative reactivity) and the SFP geometry which is designed assuming use of unborated water even though soluble boron is available (see Specification 4.3.1.1). The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents which credit use of the soluble boron may be limited to a small fraction of the total operating time.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of the NRC Policy Statement.

LCO

107
220
207

The SFP boron concentration is required to be within the ~~limit specified in the COLR~~ ~~300 ppm~~. The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 3 and 4 (i.e., a fuel handling accident). This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFP until the fuel assemblies have been verified to be stored correctly.

(continued)

BASES (continued)

APPLICABILITY This LCO applies whenever fuel assemblies are stored in the SFP, until a SFP verification has been performed following the last movement of fuel assemblies in the SFP. The SFP verification is accomplished by performing SR 3.7.13.1 or SR 3.7.13.2 after movement of fuel assemblies depending on which SFP region was affected by the fuel movement. If fuel was moved into both regions, then both SR 3.7.13.1 and SR 3.7.13.2 must be performed after the completion of fuel movement before exiting the Applicability of this LCO. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

This LCO does not apply to fuel movement within a SFP region since the accident analyses assume each region is completely filled in an infinite array.

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to immediately initiate action to perform a SFP verification (SR 3.7.13.1 and SR 3.7.13.2). The performance of this verification removes the plant from the Applicability of this LCO. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.12.1

107
200
207

This SR verifies that the concentration of boron in the SFP is within the limit ~~specified in the COLR~~. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.

This SR is required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished (i.e., SR 3.7.13.1 and 3.7.13.2).

REFERENCES

1. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
 2. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
 3. Westinghouse, "Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters," dated June 1994.
 4. UFSAR, Section 15.7.3.
-

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND

(147)

The spent fuel pool (SFP) is divided into two separate and distinct regions (see Figure B 3.7.13-1) which, for the purpose of criticality considerations, are considered as separate pools (Ref. 1). Region 1, with 176 storage positions, is designed to accommodate new or spent fuel utilizing a two of four checkerboard arrangement. A fuel assembly with an enrichment of ≤ 4.05 wt% can be stored at any available location in Region 1 since the accident analyses were performed assuming that Region 1 was filled with fuel assemblies of this enrichment. A fuel assembly with an enrichment > 4.05 wt% U-235 can also be stored in Region 1 provided that integral burnable poisons are present in the assemblies such that ~~k-infinity in the normal reactor core configuration and cold conditions~~ is ≤ 1.458 . The existing design uses Integral Fuel Burnable Absorbers (IFBAs) as the poison for fuel assemblies with enrichments > 4.05 wt%. IFBAs consist of neutron absorbing material which provides equivalencing reactivity holddown (i.e., neutron poison) that allows storage of higher enrichment fuel. The neutron absorbing material is a non-removable or integral part of the fuel assembly once it is applied. The infinite multiplication factor, K-infinity, is a reference criticality point of each fuel assembly that if maintained ≤ 1.458 , will result in a $k_{\text{eff}} \leq 0.95$ for Region 1. The K-infinity limit is derived for constant conditions of normal reactor core configuration (i.e., typical geometry of fuel assemblies in vertical position arranged in an infinite array) at cold conditions (i.e., 68°F and 14.7 psia).

Region 2, with 840 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.7.13-1, in the accompanying LCO. The storage of fuel assemblies which are within the acceptable range of Figure 3.7.13-1 in Region 2 ensures a $K_{\text{eff}} \leq 0.95$ in this region.

(continued)



BASES (continued)

BACKGROUND
(continued)

169

Consolidated rod storage canisters can also be stored in either region in the SFP provided that the minimum burnup of Figure 3.7.17-13.7.13 is met. In addition, all canisters placed into service after 1994 must have ≤ 144 rods or ≥ 256 rods (Ref. 2). The canisters are stainless steel containers which contain the fuel rods of a maximum of two fuel assemblies (i.e., 358 rods). All bowed, broken, or otherwise failed fuel rods are first stored in a stainless steel tube of 0.75 inch outer diameter before being placed in a canister. Each canister will accommodate 110 failed fuel rod tubes.

The water in the SFP normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water such that configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded.

The double contingency principle discussed in ANSI N16.1-1975 (Ref. 3) and Reference 4 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with this LCO. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 2 and 5). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by voiding (which would result in the addition of negative reactivity) and the SFP geometry which is designed assuming use of unborated water even though soluble boron is available (see Specification 4.3.1.1). The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents which credit use of the soluble boron may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the SFP ensure the k_{eff} of the SFP will always remain < 0.95 , assuming the pool to be flooded with unborated water (Specification 4.3.1.1). For fuel assemblies stored in Region 1, each assembly must have a K-infinity of ≤ 1.458 in the normal reactor core configuration at cold conditions.

~~Normal reactor core configuration is the typical geometry of fuel assemblies in the vertical position arranged in an infinite array stored in Region 2, initial enrichment and burnup shall be within the acceptable area of the Figure 3.7.13-1. Cold conditions is defined as 68°F and an atmospheric pressure of 14.7 psia.~~

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(continued)

BASES (continued)

~~LCO~~ ~~For fuel assemblies stored in Region 2, initial enrichment~~
~~(continued)~~ ~~and burnup shall be within the acceptable area of the Figure~~
~~3.7.13-1. The x-axis of Figure 3.7.13-1 is the nominal~~
U-235 enrichment wt% which does not include the ± 0.05 wt%
tolerance that is allowed for fuel manufacturing and listed
in Specification 4.3.1.1.

(continued)

BASES (continued)

APPLICABILITY This LCO applies whenever any fuel assembly is stored in the SFP.

ACTIONS

A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If ~~unable to move~~ moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

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SURVEILLANCE
REQUIREMENTS

SR 3.7.13.1

This SR verifies by administrative means that the K-infinity of each fuel assembly is ≤ 1.458 ~~in the normal reactor core configuration at cold conditions~~ prior to storage in Region 1. If the initial enrichment of a fuel assembly is ≤ 4.05 wt%, a K-infinity of ≤ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that k-infinity ≤ 1.458 ~~in the normal reactor core configuration at cold conditions~~ prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less

147

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147

(continued)

BASES (continued)

reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

(continued)

BASES (continued)

REQUIREMENTS

SURVEILLANCE SR 3.7.13.1 (continued)

Though not required for this LCO, this SR must also be performed after completion of fuel movement into Region 1 to exit the Applicability of LCO 3.7.12, "SFP Boron Concentration."

This SR is modified by a Note which states that this verification is not required when transferring a fuel assembly from Region 2 to Region 1. The verification is not required since Region 2 is the limiting SFP region, and as such, the fuel has already been verified to be acceptable for storage in Region 1.

SR 3.7.13.2

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-1 in the accompanying LCO prior to storage in Region 2. Once a fuel assembly has been verified to be within the acceptable range of Figure 3.7.13-1, further verifications are no longer required since the initial enrichment or burnup will not adversely change. For fuel assemblies in the unacceptable range of Figure 3.7.13-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

REQUIREMENTS

SURVEILLANCE ~~SR 3.7.13.2 (continued)~~

Though not required for this LCO, this SR must also be performed after completion of fuel movement into Region 2 to exit the Applicability of LCO 3.7.12.

REFERENCES

1. UFSAR, Section 9.1.2.
 2. Westinghouse, "Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters," dated June 1994.
 3. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
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(continued)

BASES (continued)

REFERENCES

(continued)

4. Letter from B.K. Grimes, NRC, to All Power Reactor—
Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
5. UFSAR, Section 15.7.3.
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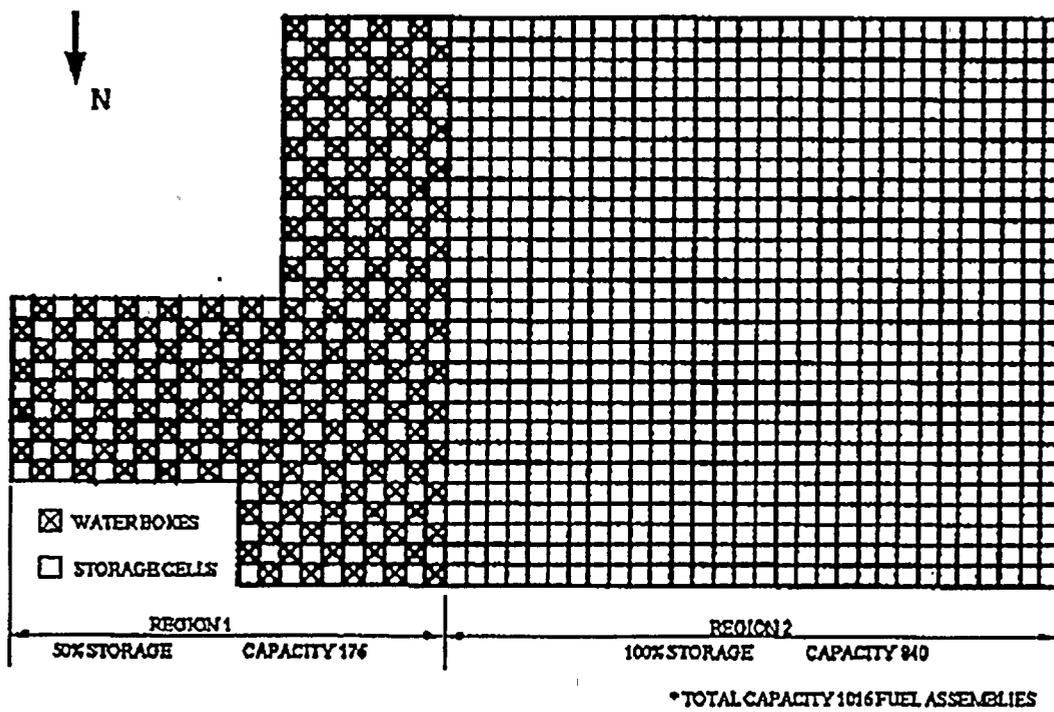
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Figure—

Spent Fuel Storage Racks



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Secondary Specific Activity

Figure B 3.7.13-1
Spent Fuel Pool



~~Spent Fuel Pool~~

B 3.7.13.1

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator (SG) tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes can be observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and Design Basis accidents (DBAs).

This limit is based on an activity value that might be expected from a 0.1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 $\mu\text{Ci/gm}$ (LCO 3.4.16, "RCS Specific Activity"). A steam line break (SLB) is assumed to result in the release of the noble gas and iodine activity contained in the SG inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be approximately 10 rem if the main steam safety valves (MSSVs) were left open for 2 hours following a trip from full power. Operating a plant at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES

The accident analysis of the SLB, (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an SLB do not exceed a small fraction of the plant EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining SG is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valve (ARV). The Auxiliary Feedwater System supplies the necessary makeup to the SG. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the SG connected to the failed steam line is assumed to be released directly to the environment within 60 seconds. The unaffected SG is assumed to discharge steam and any entrained activity through the MSSVs and ARV for the initial two hours of the event. Primary coolant was assumed to be 3.0 $\mu\text{Ci/gm}$ for this analysis based on previously allowed limits which is a factor of three greater than current limits specified in LCO 3.4.16. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The specific activity of the secondary coolant is required to be $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a DBA to a small fraction of the required limit (Ref. 1).

(continued)



BASES

LCO
(continued)

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the plant in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere from a SLB.

In MODES 5 and 6, the SGs are not being used for heat removal. Both the RCS and SGs are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 86 hours, and in MODE 5 within 4036 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
 2. Letter from D. M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: "SEP Topic, XV-2, Spectrum of Steam System Piping Failures Inside and Outside Containment; XV-12, Spectrum of Rod Ejection Accidents; XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment; XV-17, Steam Generator Tube Failure; and XV-20, Radiological Consequences of Fuel Damaging Accidents - R.E. Ginna," dated September 24, 1981.
-

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - MODES 1, 2, 3, and 4

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

- a. One qualified independent offsite power source-circuit connected between the offsite transmission network and each of the onsite 480 V safeguards buses required by LCO 3.8.9, "Distribution Subsystems - MODES 1, 2, 3, and 4"; and
- b. Two emergency diesel generators (DGs) capable of supplying their required respective onsite 480 V safeguards buses required by LCO 3.8.9.

212

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No offsite Offsite power to one or more 480 V safeguards bus(es) inoperable.	A.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition A concurrent with inoperability of redundant required feature(s)
	AND A.2 Restore offsite circuit to OPERABLE status.	72 hours

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One DG inoperable. (212)	B.1 Perform SR 3.8.1.1 for the offsite circuit.	1 hour <u>AND</u> Once per 128 hours thereafter
	<u>AND</u> B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	<u>AND</u> B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	<u>OR</u> B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	<u>AND</u> B.4 Restore DG to OPERABLE status.	7 days

(continued)



ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>212</p> <p>C. No offsite Offsite power to one or more 480 V safeguards bus(es) inoperable.</p> <p><u>AND</u></p> <p>One DG inoperable.</p>	<p>-----NOTE-----</p> <p>Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4," when Condition C is entered with no AC power source to one distribution train.</p> <p>-----</p> <p>C.1 Restore required offsite circuit to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2 Restore DG to OPERABLE status.</p>	<p>12 hours</p> <p>12 hours</p>
<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
<p>E. Two DGs inoperable.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for the offsite circuit to each of the 480 V safeguards buses.	7 days
SR 3.8.1.2 -----NOTES----- 1. Performance of SR 3.8.1.9 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. ----- Verify each DG starts from standby conditions and achieves rated voltage and frequency.	31 days
SR 3.8.1.3 -----NOTES----- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.9. ----- Verify each DG is synchronized and loaded and operates for ≥ 60 minutes and < 120 minutes at a load ≥ 1950 kW and < 2250 kW.	31 days

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(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.4 Verify the fuel oil level in each day tank.	31 days
(continued) SR 3.8.1.5 Verify the DG fuel oil transfer system operates to transfer fuel oil from each storage tank to the associated day tank.	31 days
<p>SR 3.8.1.6 Verify transfer of AC power sources from the preferred offsite circuit (50/50 mode) 50/50 mode to the alternate offsite circuit (100/0/100/0 mode and 0/100 mode) mode.</p> <p>212</p>	24 months
<p>SR 3.8.1.7 ----- NOTE ----- ----- NOTES ----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.</p> <p>----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.</p> <p>232</p> <p>2. Credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify each DG does not trip during and following a load rejection of ≥ 295 kW.</p>	24 months
<p>SR 3.8.1.8 ----- NOTE ----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4.</p>	(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.8 NOTES</p> <p>1. This Surveillance shall not be performed in MODE 1, 2, 3, or 4.</p> <p>2. Credit may be taken for unplanned events that satisfy this SR.</p> <p>Verify each DG's automatic trips are bypassed on an actual or simulated safety injection (SI) signal except:</p> <ul style="list-style-type: none"> a. Engine overspeed; b. Low lube oil pressure; and c. Start failure (overcrank) relay. 	<p>24 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. <hr/> <p>3. Credit may be taken for unplanned events that satisfy this SR.</p> <hr/> <p>Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal:</p> <ol style="list-style-type: none"> a. De-energization of 480 V safeguards buses; b. Load shedding from 480 V safeguards buses; and c. DG auto-starts from standby condition energizes automatically connected emergency loads, and operates for ≥ 5 minutes. <hr/> <ol style="list-style-type: none"> 1. energizes permanently connected loads; 2. energizes auto-connected emergency loads through the load sequencer, and 3. supplies permanently and auto-connected emergency loads for ≥ 5 minutes. 	<p>24 months</p>

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SURVEILLANCE REQUIREMENTS
3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - MODES 5 and 6

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

- (212)
- a. One qualified independent offsite power source-circuit connected between the offsite transmission network and each of the onsite 480 V safeguard buses required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6"; and
 - b. One emergency diesel generator (DG) capable of supplying one train of the onsite 480 V safeguard bus(es) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------	-----------------	-----------------

A. ~~No offsite~~ Offsite power to one or more required 480 V safeguards bus(es) inoperable.

212

-----NOTE-----
Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A.

A.1 Declare affected required feature(s) with ~~no offsite power available from a required circuit~~ inoperable.

Immediately

OR

A.2.1 Suspend CORE ALTERATIONS.

Immediately

AND

~~A.2.2 Suspend movement of irradiated fuel assemblies.~~

AND

(continued)

A. (continued)

~~A.2.2.3~~
Initiate action to suspend operations involving positive reactivity additions.

Immediately

AND

~~A.2.3.4~~
Initiate action to restore required offsite power circuit to OPERABLE status.

Immediately

ACTIONS

B. No-DG to the required
480 V safeguards
bus(es) inoperable.

212

B.1 Suspend CORE
ALTERATIONS. Immediately

AND

B.2 ~~Initiate action to
suspend operations
involving positive
reactivity~~ Immediately

~~additions Suspend
movement of
irradiated fuel
assemblies.~~

~~Immediately~~

AND

B.3 ~~Initiate action to
restore required DG
to OPERABLE~~ Immediately

~~status suspend
operations involving
positive reactivity
additions.~~

AND

B.4 ~~Initiate action to
restore required DG
to OPERABLE status.~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	For AC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.1.1 SR 3.8.1.4 SR 3.8.1.2 SR 3.8.1.5	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil

LCO 3.8.3 The stored diesel fuel oil shall be within limits for each required emergency diesel generator (DG).

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2,
"AC Sources - MODES 5 and 6."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DGs with onsite fuel oil supply not within limit.	A.1 Restore fuel oil level to within limit.	48 ¹² hours
B. One or more required DGs with stored fuel oil total particulates not within limit.	B.1 Restore fuel oil total particulates within limit.	7 days
C. One or more DGs with new fuel oil properties not within limits.	C.1 Restore stored fuel oil properties within limits.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>212 One or more required DGs diesel fuel oil not within limits for reasons other than Condition A or B, or C.</p>	<p>CD.1 Declare associated DG inoperable.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.3.1 Verify ^{each fuel oil storage tank contains} an on-site supply of ≥ 5000 gal of diesel fuel oil available for each required DG.</p> <p>169</p>	<p>31 days</p>
<p>SR 3.8.3.2 Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.</p> <p>212</p>	<p>In accordance with the Diesel Fuel Oil Testing Program</p>



3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - MODES 1, 2, 3, and 4

LCO 3.8.4 The Train A and Train B DC electrical power sources shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power source inoperable.	A.1 Restore DC electrical power source to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in Mode 5.	36 hours
C. Both DC electrical power sources inoperable.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.1 234 Verify battery charging capacity after terminal voltage is ≥ 150 amps 129 V on float charge.</p>	<p>317 days</p>
<p>SR 3.8.4.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. <p>-----</p> <p>Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.</p>	<p>24 months</p>

(continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.3 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -----</p> <p>Verify battery capacity is \geq 80% of the manufacturer's rating when subjected to a performance discharge test.</p>	<p>60 months</p> <p><u>AND</u></p> <p>12 months when battery shows degradation, or has reached 85% of expected life with capacity < 100% of manufacturer's rating</p> <p><u>AND</u></p> <p>24 months when battery has reached 85% of the expected life with capacity \geq 100% of manufacturer's rating</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - MODES 5 and 6

LCO 3.8.5 DC electrical power sources shall be OPERABLE to support the DC electrical power distribution subsystem required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----------	-----------------	-----------------



A. ~~A~~ One or more required DC electrical power source(s) inoperable.

212

A.1 Declare affected required feature(s) inoperable. Immediately

OR

A.2.1 Suspend CORE ALTERATIONS. Immediately

AND

~~A.3 Restore battery cell parameters to Category A and B limits 2.2~~ Immediately

~~Suspend movement of Table B 3.8.6 Irradiated fuel assemblies.~~ Immediately

~~B.~~

~~Required Action and associated Completion Time of Condition A not met.~~ Immediately

OR

~~One or more batteries with average electrolyte temperature of the representative cells < 65°F.~~

OR

~~One or more batteries with one or more battery cell parameters not within Category C values.~~
(continued)

1-hour

24-hours

ACTIONS

ACTIONS
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.5.1 For DC sources required to be OPERABLE, the following SR is applicable. <u>169</u> <u>SR 3.8.4.1</u>	In accordance with applicable SR

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B batteries shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DC electrical power sources are required to be OPERABLE by LCO 3.8.5, "DC Sources -MODES 5 and 6."

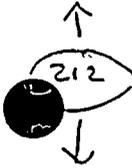
ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more batteries with one or more battery cell parameters not within Category A or B limits.</p> <p>212 A. One or more batteries with one or more battery cell parameters not within limits.</p>	<p>A.1 Verify pilot cells electrolyte level and float voltage meet Table B-3.8.6-1 Category C limits.</p> <p><u>AND</u></p> <p>A.2 Verify battery cell parameters meet Table B-3.8.6-1 Category C limits.</p> <p>1 Declare associated battery inoperable.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.6.1 Verify electrolyte level of each connected battery cell parameters meet Table B 3.8.6-1 Category A limits is above the top of the plates and not overflowing.	31 days
SR 3.8.6.2 Verify the float voltage of each connected battery cell parameters meet Table B 3.8.6-1 Category B limits is > 2.07 V.	92 days AND Once within 7 days after a battery discharge < 105 V AND Once within 7 days after a battery overcharge > 150 V 31 days
SR 3.8.6.3 Verify average electrolyte temperature specific gravity of representative cells the designated pilot cell in each battery is $\geq 65^{\circ}\text{F}$ 1.188 for Battery A and ≥ 1.192 for Battery B.	92 31 days
SR 3.8.6.4 Verify average electrolyte temperature of the designated pilot cell in each battery is $\geq 55^{\circ}\text{F}$.	31 days
SR 3.8.6.5 Verify average electrolyte temperature of every fifth cell of each battery is $\geq 55^{\circ}\text{F}$.	92 days



(2)

(2)

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.6.6 Verify specific gravity of each connected battery cell is:</p> <ul style="list-style-type: none"> a. Not more than 0.020 below average of all connected cells, and b. Average of all connected cells is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. 	<p>92 days</p>

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3.8 ELECTRICAL POWER SYSTEMS

3.8.7 AC Instrument Bus Sources - MODES 1, 2, 3, and 4

LCO 3.8.7 The following AC instrument bus power sources shall be OPERABLE:

- a. Inverters for Instrument Buses A and C; and
- b. Class 1E constant voltage transformer (CVT) for Instrument Bus B.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One inverter inoperable.	A.1 Power AC instrument bus from its Class 1E or non-Class 1E CVT.	2 hours
	<u>AND</u>	
	A.2 Power AC instrument bus from its Class 1E CVT.	24 hours
	<u>AND</u>	
	A.3 Restore inverter to OPERABLE status.	72 hours
B. Class 1E CVT for AC Instrument Bus B inoperable.	B.1 Power AC Instrument Bus B from its non-Class 1E CVT.	2 hours
	<u>AND</u>	
	B.2 Restore Class 1E CVT for AC Instrument Bus B to OPERABLE status.	7 days

AC Instrument Bus Sources - MODES 1, 2, 3, and 4
3.8.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)		
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours
D. Two or more required instrument bus sources inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.7.1 Verify correct static switch alignment to Instrument Bus A and C.	7 days
SR 3.8.7.2 Verify correct Class 1E CVT alignment to Instrument Bus B.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 AC Instrument Bus Sources - MODES 5 and 6

LCO 3.8.8 AC instrument bus power sources shall be OPERABLE to support the onsite Class 1E AC instrument bus electrical power distribution subsystem required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
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A. One or more required AC instrument bus power source(s) inoperable.

~~A.1 Declare affected required feature(s) inoperable.~~

Immediately

OR

~~A.2.1 Suspend CORE ALTERATIONS. A.1 Declare affected required feature(s) inoperable.~~

Immediately

Immediately

OR

~~A.2.1 Suspend CORE ALTERATIONS.~~

~~Immediately~~

AND

~~A.2.2 Suspend movement of irradiated fuel assemblies.~~

AND

A.2.3 Initiate action to suspend operations involving positive reactivity additions.

Immediately

AND

A.2.4 Initiate action to restore required AC instrument bus power source(s) to OPERABLE status.

212

D
ACTIONS
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct static switch alignment to required AC instrument bus(es).	7 days
SR 3.8.8.2	Verify correct Class 1E CVT alignment to the required AC instrument bus.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - MODES 1, 2, 3, and 4

LCO 3.8.9 Train A and Train B of the following electrical power distribution subsystems shall be OPERABLE:

- a. AC power;
- b. AC instrument bus power; and
- c. DC power.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution train inoperable.	A.1 Restore AC electrical power distribution train to OPERABLE status.	8 hours
B. One required AC instrument bus electrical power distribution train inoperable.	B.1 Restore required AC instrument bus electrical power distribution train to OPERABLE status.	2 hours
C. One DC electrical power distribution train inoperable.	C.1 Restore DC electrical power distribution train to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Conditions A, B, or C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours
212 E. Two or more distribution trains with inoperable electrical power distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
212 SR 3.8.9.1 Verify correct breaker alignments and voltage to required AC, DC, and AG instrument bus electrical power distribution subsystems trains.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - MODES 5 and 6

LCO 3.8.10 The necessary ~~portion~~ ~~trains (S)~~ of AC, DC, and AC instrument bus ~~the following~~ electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

a.

APPLICABILITY: ~~MODES 5 and 6~~ AC power.

b.

ACTIONS

AC instrument bus power; and

c. DC power.

APPLICABILITY: MODES 5 and 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
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212
↓



212

<p>A. One or more required AC, DC, or AC instrument bus electrical power distribution train train(s) inoperable.</p>	<p>A.1 Declare associated supported required feature(s) inoperable.</p> <p><u>OR</u></p> <p>A.2.1 Suspend CORE ALTERATIONS.</p> <p><u>AND</u></p> <p>A.2.2 Suspend movement of irradiated fuel assemblies.</p> <p><u>AND</u></p> <p>A.2.3 Initiate action to suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p> <p>Immediately</p> <p>(continued)</p>
<p>A. (continued)</p>	<p>A.2.4 Initiate actions to restore required AC, DC, and AC instrument bus electrical power distribution train(s) to OPERABLE status.</p> <p><u>AND</u></p> <p>A.2.5 Declare associated required residual heat removal loop(s) inoperable and not in operation.</p>	<p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
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D
ACTIONS

SR 3.8.10.1 Verify correct breaker alignments and
voltage to required AC, ~~DC,~~ and ~~AC~~
~~instrument bus electrical power~~
distribution subsystem ~~strains~~.

7 days

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential active components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for each of these electrical power sources which are further divided and organized based on voltage considerations and safety classification. This LCO is provided to specify the minimum sources of AC power which are required to supply the 480 V safeguards buses and associated distribution subsystem during MODES 1, 2, 3, and 4.

The plant AC sources consist of an independent offsite power source and the onsite standby emergency power source (Ref. 1). Atomic Industrial Forum (AIF) GDC 39 (Ref. 2) requires emergency power sources be provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning of the Engineered Safety Features (ESF) and protection systems. The offsite and onsite AC sources can each supply power to 480 V safeguards buses to ensure that reliable power is available during any normal or emergency mode of plant operation. The 480 V safeguards buses are divided into redundant trains so that the loss of any one train does not prevent the minimum safety functions from being performed. Safeguards Buses 14 and 18 are associated with Train A and safeguards Buses 16 and 17 are associated with Train B. Since only the onsite standby power source is classified as Class 1E, the offsite power source is not required to be separated into redundant trains.

(continued)

BASES

(continued)

BASES

BACKGROUND
—(continued)

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The independent offsite power source consists of breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite 480 V safeguards buses. The independent offsite power source essentially begins from two station auxiliary transformers (SAT 12A and 12B) each supplied from an independent transmission line emanating from separate switchyards (see Figure B 3.8.1-1). SAT 12A is connected to the 34.5 kV transmission system (circuit 751) and SAT 12B is connected to the plant 115 kV switchyard (circuit 767). The SATs may be configured in the following modes:

- a. SAT 12A (or SAT 12B) supplies safeguards Buses 16 and 17 and SAT 12B (or SAT 12A) supplies safeguards Buses 14 and 18 (50/50 mode);
- b. SAT 12A supplies all safeguards Buses (0/100 mode); or
- c. SAT 12B supplies all safeguards Buses (100/0 mode).

The preferred configuration is the 50/50 mode; however, all three modes of operation meet applicable design requirements for normal operation (Ref. 1). Offsite power can also be provided during an emergency through the plant auxiliary transformer 11 by backfeeding from the 115 kV transmission system and main transformer.

SATs 12A and 12B are each connected to two non-Class 1E, 4.16 kV buses (12A and 12B). The 4.16 kV Bus 12A feeds the Class 1E loads on the 480 V safeguards Buses 14 and 18 and 4.16 kV Bus 12B feeds the Class 1E loads on the 480 V safeguards Buses 16 and 17 (see Figure B 3.8.1-1). Loss of power to any of the safeguards buses, as a result of inoperable offsite circuit component(s), is a loss of offsite power. The offsite power source ends after the feeder breaker supplying each 480 V safeguards bus.

(continued)

BASES

(continued)

BASES

BACKGROUND
(continued)

The onsite standby power sources consist of two 1950 kW continuous rating emergency diesel generators (DGs) connected to the safeguards buses to supply emergency power in the event of loss of all other AC power. The DGs are located in separate rooms in a Seismic Category I structure located adjacent to the northeast wall of the Turbine Building. Each DG room has its own ventilation system. The ventilation system is designed to maintain the DG room between 60°F and 104°F and to remove any hydrocarbon gases in the room (Ref. 3). Each ventilation system consists of two fans and associated ductwork and dampers that fail open on loss of instrument air and control power. One fan is designed to start on DG actuation with a second fan designed to start when the room temperature reaches 90°F. The second fan's discharge air flow is directed to the DG control panel and has a delayed start to prevent potentially freezing the cooling water jacket piping during cold weather conditions.

The DGs utilize an air motor for starting. The air motor is supplied by two receivers which provide sufficient air for five DG starts before requiring a recharge of the receivers. The DGs are supplied by separate fuel oil day tanks which can be cross-tied if required. Additional fuel oil can be transferred from redundant underground fuel oil storage tanks. A dedicated fuel oil transfer pump is used for this transfer. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank, to result in the loss of more than one DG.

DG A is dedicated to safeguards Buses 14 and 18 and DG B is dedicated to safeguards Buses 16 and 17. A DG starts automatically on a safety injection (SI) signal or on an undervoltage signal on its corresponding 480 V buses (refer to LCO 3-3-53 3.4, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). In the event of only an SI signal, the DGs automatically start and operate in the standby mode without tying to the safeguards buses.

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(continued)

BASES

(continued)



BASES

BACKGROUND
(continued)

In the event of loss of offsite power, or abnormal offsite power where offsite power is tripped as a consequence of bus undervoltage or degraded voltage, the DGs automatically start and tie to their respective buses. All bus loads except for the containment spray (CS) pump, component cooling water (CCW) pump and safety related motor control centers are tripped upon actuation of the undervoltage relays. This is independent of or coincident with an SI signal. Once the undervoltage relay resets independent of a SI signal, the operator may manually connect loads onto the bus(es). During a coincident SI signal, the CCW pump is also tripped and loads are sequentially connected to their respective buses by the automatic load sequencer.

In the event of loss of offsite power to only one safeguards bus in a train, the DG will automatically start and tie only to the affected bus. During a coincident SI signal, the normal feed breaker on the second bus on the affected train will be tripped by the undervoltage relay on the failed bus causing the DG to automatically tie to both buses. This condition will then actuate the automatic load sequencer.

In the event of a loss of offsite power and a coincident SI signal, the electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA). Certain required plant loads are returned to service in a predetermined sequence by the automatic load sequencer in order to prevent overloading the DG during the start process. Within approximately 1 minute after the initiating signal is received, all loads needed to recover the plant or maintain it in a safe condition are returned to service.

(continued)

BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The initial conditions of DBA and transient analyses (Refs. 4 and 5) assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This results in maintaining at least one train of the onsite standby power or offsite AC sources OPERABLE in the event of:

- a. An assumed loss of all AC offsite power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC electrical power sources ensures that one train of offsite or onsite standby AC power is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one train of onsite standby power).

(continued)

BASES

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of offsite power also ensures that at least one AC power source is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 1). Therefore, the requirements of GDC 17 (Ref. 6) can be met at all times.

The DGs are designed to operate following a DBA or anticipated operational occurrence (AOO) until offsite power can be restored. An AOO is defined as a Condition 2 event in Reference 7 (i.e., events which can be expected to occur during a calendar year with moderate frequency). The DGs are required to start within 10 seconds and begin loading. The DGs can begin receiving up to 30% of design loads after the 10 second start time and can accept 100% of design loads within 30 seconds. The DGs are manually loaded if only an undervoltage signal is present and load sequenced if a coincident undervoltage and SI signal is present. The loads are sequenced as follows (assume SI signal at 0 seconds):

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	DG Load	DG A Time	DG B Time
	480V safeguards buses and CS pumps	10	10
	SI pump A and B	15	15
	SI pump C	20	22
	Residual heat removal pump	25	27
	Selected service water pump	30	32
	First containment recirculation fan cooler	35	37
	Second containment recirculation fan cooler	40	42
	Motor driven auxiliary feedwater pump	45	47

Since the DGs must start and begin loading within 10 seconds, only one air start must be available in the air receivers as assumed in the accident analyses. The long term operation of the DGs (until offsite power is restored) is discussed in LCO 3.8.3, "Diesel Fuel Oil."

The AC sources satisfy Criterion 3 of NRC Policy Statement.

LCO

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One qualified independent offsite power source circuit connected between the offsite transmission network and the onsite 480 V safeguards buses and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA.

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An OPERABLE qualified independent offsite power source circuit is one that is capable of maintaining rated voltage, and accepting required loads during an accident, while connected to the 480 V safeguards buses required by LCO 3.8.9, "Distribution Subsystems - MODES 1, 2, 3, and 4."

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Power from either offsite power circuit 751 or 767 satisfies this requirement.

A DG is considered OPERABLE when:

- a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;

(continued)



BASES

LCO
(continued)

- b. All loads on each 480 V safeguards bus except for the safety related motor control centers, CCW pump, and CS pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident SI and undervoltage signal);
- c. The DG is capable of accepting required loads both manually and within the assumed loading sequence intervals following a coincident SI and undervoltage signal, and continue to operate until offsite power can be restored to the safeguards bus (i.e., 40 hours);
- d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;
- e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and
- f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

The AC sources in one train must be separate and independent of the AC sources in the other train. For the DGs, separation and independence must be complete assuming a single active failure. For the independent offsite power source, separation and independence are to the extent practical (i.e., operation is preferred in the 50/50 mode, but may also exist in the 100/0 or 0/100 mode).

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded, as a result of AOOs or abnormal transients; and

(continued)

BASES

APPLICABILITY
(continued)

- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - MODES 5 and 6."

ACTIONS

A.1 and A.2

(212)

With ~~no~~ offsite power to one or more 480V safeguard bus(es) inoperable, assurance must be provided that a coincident single failure will not result in a complete loss of required safety features. If the redundant safety feature to the component or train affected by the loss of offsite power is also unavailable, the assumption that two complete safety trains are OPERABLE may no longer exist. As an example, if offsite power were unavailable to 480 V Bus 14, DG A could supply the necessary power to the bus. If residual heat removal pump (RHR) B (supplied power by Bus 16) were inoperable at the same time, or at any time after the loss of offsite power to Bus 14, a loss of redundant required safety features exists since a failure of DG A would result in the loss of emergency core cooling. Therefore, RHR pump A on Bus 14 would have to be declared inoperable within 12 hours after RHR pump B and offsite power to Bus 14 were declared unavailable.

The Completion Time of 12 hours as provided by Required Action A.1 to declare the required safety features inoperable is based on the fact that it is less than the Completion Time for restoring OPERABILITY of the offsite power circuit and all safety features affected by the loss of the 480 V bus. A shorter Completion Time is provided since the required safety features have been potentially degraded by the loss of offsite power (i.e., using the same example as above, the 72 hour Completion Time for restoring RHR pump B was developed assuming that RHR pump A had both offsite and onsite standby emergency power available). Therefore, a penalty is assessed to only allow 12 hours in this configuration.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The Completion Time for Required Action A.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time is an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that:

- a. There is no offsite power available to one or more 480 V safeguards bus; and
- b. A redundant required feature is inoperable on a second 480 V safeguards bus.

If at any time during the existence of Condition A, a redundant required feature becomes inoperable, this Completion Time begins to be tracked. Required Action A.1 can be exited if the inoperable DG or the required feature on the OPERABLE DG is restored to OPERABLE status.

The level of degradation during Condition A means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite standby AC sources have not been degraded. This level of degradation generally corresponds to either:

- a. Loss of offsite power sources to SAT 12A and/or SAT 12B;
- b. Failure of SAT 12A or 12B or 4.16 kV Bus 12A or 12B;
or
- c. Failure of a station service transformer supplying a 480 V safeguards bus.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

With a total loss of the offsite power sources to SAT 12A and 12B, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident for either train. With loss of offsite power to SAT 12A or 12B, failure of SAT 12A or 12B, or failure of Bus 12A or 12B, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident for a single AC electrical train. With a failure of a station service transformer, the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the consequences of an accident for one 480 V safeguards bus in one AC electrical train. In all cases, sufficient onsite AC sources are available to maintain the plant in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 72 hour Completion Time provides a period of time to effect restoration of the offsite circuit commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

B.1

With one DG inoperable, it is necessary to verify the availability of the offsite circuit to each of the affected 480 V safeguards buses on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met (i.e., Condition D would not apply). However, if a circuit fails to pass SR 3.8.1.1, it is inoperable and Condition C would be entered.

(continued)

BASES

213 The Completion Time of 1 hour to perform SR 3.8.1.1 is based on the importance of this verification to ensure that offsite power is available to the affected bus. The Frequency of once per 8 hours thereafter is based on the alarms and indications of breaker status that are available in the control room.

(continued)

BASES

ACTIONS
(continued)

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of a safety feature. These features are designed with redundant safety related trains which are supplied power from separate and independent onsite power sources. If one onsite power source is inoperable, it must be assured that the redundant safety related train supplied by the OPERABLE DG is available to provide the necessary safety function.

The Completion Time of 4 hours for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time is an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. An inoperable DG exists; and
- b. A required feature on the other train (Train A or Train B) is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature supported by the OPERABLE DG subsequently becomes inoperable, this Completion Time would begin to be tracked. Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are supplied power by the OPERABLE DG, results in starting the Completion Time for Required Action B.2. In this Condition, the remaining OPERABLE DG and the offsite circuit are adequate to supply electrical power to the onsite 480 V safeguards buses.

(continued)

BASES

ACTIONS

B.2 (continued)

The Completion Time of 4 hours to declare the required safety features inoperable is based on the fact that it is less than the Completion Time for restoring OPERABILITY of the DG and all safety features supported by the DG. A shorter Completion Time is provided since the required safety features have been potentially degraded by the inoperable DG. Therefore, a penalty is assessed to only allow 4 hours in this configuration. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Required Action B.2 can be exited if the inoperable DG or the required feature on the OPERABLE DG is restored to OPERABLE status.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined within 24 hours that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 is not required to be performed. If the cause of inoperability is determined to exist on the other DG, the second DG would be declared inoperable upon discovery and Condition E would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the second DG within 24 hours, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, activities must continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

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ACTIONS — ~~The 24 hour Completion Time is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG (Ref. 3.1 and BB).~~

(continued)

BASES

3.2 (continued)

~~The 24 hour Completion Time is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG (Ref. ACTIONS B.8)4~~
(continued)

(104)

~~With one inoperable DG, the remaining OPERABLE DG and the offsite circuit are adequate to supply electrical power to the onsite 480 V safeguards buses.~~

B.4

(212)

~~With one inoperable DG, the remaining OPERABLE DG and the offsite circuit are adequate to supply electrical power to the onsite 480 V safeguards buses. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.~~

C.1

With no offsite power to one or more 480 V safeguards bus(es) and one DG inoperable, redundancy is lost in both the offsite and onsite AC electrical power systems. Since power system redundancy is provided by these two diverse sources of power, the AC power sources are only degraded and no loss of safety function has occurred since at least one DG and potentially one offsite AC power source are available. However, the plant is vulnerable to a single failure which could result in the loss of multiple safety functions. Therefore, a Completion Time of 12 hours is provided to either restore the offsite power circuit or the DG to OPERABLE status. This Completion Time is less than that for an inoperable offsite power source or an inoperable DG due to the single failure vulnerability of this configuration.

(continued)

BASES

~~As discussed in LCO 3.0.6, the AC electrical power distribution subsystem ACTIONS—C would not be entered even if all AC sources to either train were inoperable, resulting in de-energization.1 (continued)~~

~~As discussed in LCO 3.0.6, the AC electrical power distribution subsystem ACTIONS would not be entered even if all AC sources to either train were inoperable, resulting in de-energization.~~ Therefore, the Required Actions of this Condition are modified by a Note which states that the Required Actions of LCO 3.8.9, "Distribution Systems -MODES 1, 2, 3, and 4" must also be immediately entered with no AC power source to one distribution train. This allows Condition C to provide requirements for the loss of an offsite power circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

(continued)

BASES

(continued)

ACTIONS D.1 and D.2

169
If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If both DGs are inoperable, a loss of safety function would exist if offsite power were unavailable; therefore, LCO 3.0.3 must be entered.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function (Ref. 2). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions).

SR 3.8.1.1

This SR ensures proper circuit continuity for the independent offsite power source to each of the onsite 480 V safeguards buses and availability of offsite AC electrical power. Checking breaker alignment and indicated power availability verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their qualified power source. The Frequency of 7 days is adequate since breaker position is not likely to change without the operators knowledge and because alarms and indications of breaker status are available in the control room.

(continued)

BASES

(continued)

BASES

SURVEILLANCE

SR 3.8.1.2

(continued)

REQUIREMENTS

This SR verifies that each DG starts from standby conditions and achieves rated voltage and frequency. This ensures the availability of the DG to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition. The DG voltage control may be either in manual or automatic during the performance of this SR. The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

This SR is modified by two Notes. Note 1 indicates that performance of SR 3.8.1.9 satisfies this SR since SR 3.8.1.9 is a complete test of the DG. The second Note states that all DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. This minimizes the wear on moving parts that do not get lubricated when the engine is not running.

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SURVEILLANCE — SR 3.8.1.3

REQUIREMENTS

(continued)

This SR verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures. A maximum run time not to exceed 120 minutes minimizes the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.85 lagging and 0.95 lagging. The upper load band limit of 2250 kW is provided to avoid routine overloading of the DG which may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The lower load band limit is the expected maximum load following a DBA.

In addition to verifying the DG capability for synchronizing with the offsite electrical system and accepting loads, the

(continued)

BASES

DG ventilation system should also be verified during this surveillance.

(continued)

BASES

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REQUIREMENTS
(continued)

SURVEILLANCE SR 3.8.1.3

The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

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~~This SR is modified by four Notes.~~ Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients outside the load band (e.g., due to changing bus loads), do not invalidate this test. Similarly, momentary power factor transients above or below the administrative limit do not invalidate the test. Note 3 indicates that this Surveillance ~~should~~ shall be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations.

(continued)

BASES

~~SURVEILLANCE~~ ~~SR 3.8.1.4~~
~~REQUIREMENTS~~

~~(continued)~~

(212)

~~This SR provides verification that the level~~ Note 4
~~stipulates a prerequisite requirement for performance of~~
~~fuel oil in each day tank is at or above the level at which~~
~~fuel oil is automatically added when the fuel oil transfer~~
~~pump is in auto and the DG is operating.~~ this SR. A
~~successful performance of SR 3.8.1.2 or SR 3.8.1.9 must~~
~~precede this surveillance to prevent unnecessary starts of~~
~~the DGs.~~

SR 3.8.1.4

This SR provides verification that the level of fuel oil in each day tank is at or above the level at which fuel oil is automatically added when the fuel oil transfer pump is in auto and the DG is operating. This level ensures adequate fuel oil for a minimum of 1 hour of DG operation at 110% of full load. This is equivalent to a day tank level of 8.25 inches above the tank suction line.

The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

SURVEILLANCE

SR 3.8.1.5

(169)
(continued)

REQUIREMENTS

This SR demonstrates that each DG fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of the DGs. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic or manual fuel transfer systems are OPERABLE.

The Frequency of 31 days is adequate to provide assurance of DG OPERABILITY, since the design of the fuel oil transfer system is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG operation.

(continued)

BASES

SURVEILLANCE — SR 3.8.1.6

REQUIREMENTS

(continued)

(212)

This SR involves the transfer of the 480 V safeguards bus power supply from the preferred offsite power circuit configuration (50/50 mode) to the alternate offsite power circuit configurations (100/0/50/50 mode to the 100/0 mode and 0/100 mode) mode which demonstrates the OPERABILITY of the alternate circuit distribution network to power the required loads. The Frequency of 24 months is based on engineering judgment, taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE

SR 3.8.1.7

(continued)

REQUIREMENTS

This SR verifies that each DG does not trip during and following a load rejection of ≥ 295 kW. Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This SR demonstrates the DG load response characteristics and capability to reject the largest single load on the buses supplied by the DG (i.e., a safety injection pump).

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ 0.9 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

SURVEILLANCE — SR 3.8.1.7 (continued)

REQUIREMENTS

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This SR is modified by a Note stating that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems.

SR 3.8.1.8

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This SR demonstrates that DG noncritical protective functions. The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

SURVEILLANCE SR 3.8.1.8
REQUIREMENTS

(continued)

169

This SR demonstrates that DG noncritical protective functions (e.g., overcurrent, reverse power, local stop pushbutton) are bypassed on an actual or simulated SI actuation signal, and critical protective functions (engine overspeed, low lube oil pressure, and start failure (overcrank) relay) trip the DG to avert substantial damage to the DG. The noncritical trips are bypassed during DBAs but still provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The Frequency of 24 months is based on engineering judgment, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

232

This SR is modified by a Note stating that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The first Note states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that performing the Surveillance would remove a required DG from service which is undesirable in these MODES.

(continued)

BASES

232 SURVEILLANCE — ~~The second Note acknowledges that credit may be taken for unplanned events that satisfy this SR-3.8.1.9~~

REQUIREMENTS
(continued) — ~~In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.~~

~~This SR demonstrates the DG operation during an actual or simulated loss of offsite power signal in conjunction 3.8.1.9~~

~~In the event of a DBA coincident with an actual or simulated SI actuation signal a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.~~

(continued)

BASES

~~SURVEILLANCE~~ SR 3.8.1.9
~~REQUIREMENTS~~

(continued)

169 This SR demonstrates the DG operation during an actual or simulated loss of offsite power signal in conjunction with an actual or simulated SI actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months is based on engineering judgement, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

~~THE FOLLOWING TEXT WAS MOVED~~

~~This SR is modified by two Notes.~~ This SR is modified by three Notes.

~~THE PRECEDING TEXT WAS MOVED~~

Note 1 states that all DG starts may be preceded by an engine prelube period which is intended to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine lube oil continuously circulated and temperature maintained consistent with manufacturer recommendations for the DGs. Note 2 states that this Surveillance shall not be performed in MODE 1, 2, 3, or 4 since performing the Surveillance during these MODES would remove a required offsite circuit from service, cause perturbations to the electrical distribution systems, and challenge safety systems.

(continued)

BASES

232 Note 3 acknowledges that credit may be taken for unplanned events that satisfy this SR.

(continued)

BASES

REFERENCES

1. UFSAR, Chapter 8.
 2. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
 3. UFSAR, Section 9.4.9.5..
 4. UFSAR, Chapter 6.
 5. UFSAR, Chapter 15.
 6. 10 CFR 50, Appendix A, GDC 17.
 7. "American National Standard, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 8. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
-

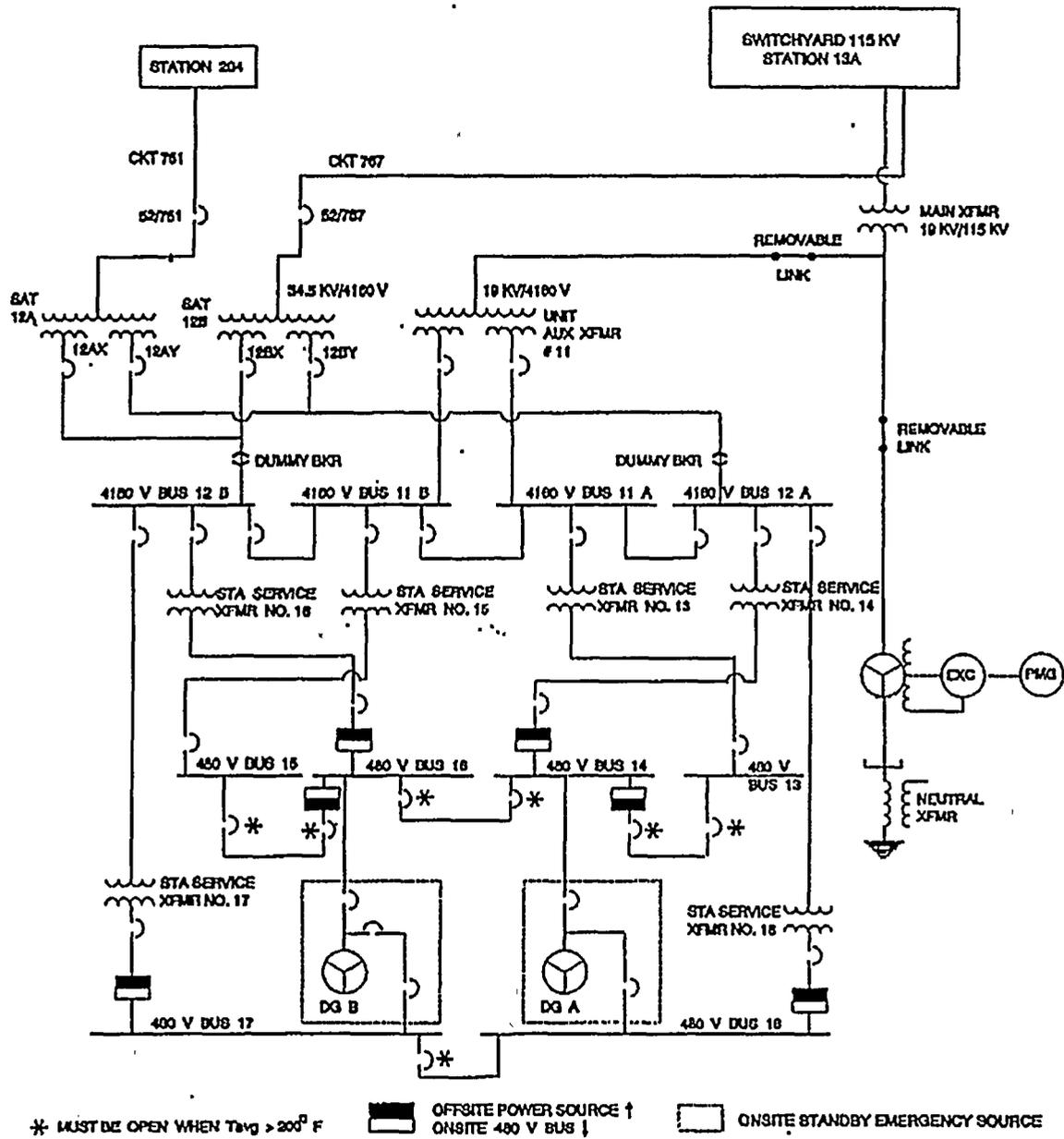


Figure B 3.8.1-1

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section for Bases 3.8.1, "AC Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6 the minimum required AC sources may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the AC power sources, must be removed from service. The minimum required AC sources is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC electrical power sources during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available; and
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available;

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC electrical power sources ensures that one train of the onsite power or offsite AC sources are OPERABLE in the event of:

(continued)



BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

(continued)

BASES

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6 this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One qualified independent offsite power circuit supplying the associated AC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10, "Distribution Systems—MODES 5 and 6," ensures that all required loads are powered from offsite power. An OPERABLE DG, capable of supporting the distribution system required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the independent offsite power circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

An OPERABLE qualified offsite circuit is one that is capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the 480 V safeguards bus(es). Power from either offsite power circuit 751 or 767, or by backfeeding through auxiliary transformer 11 satisfies this requirement.

(continued)

BASES

(continued)

BASES

LCO
(continued)

A DG is considered OPERABLE when:

- a. The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on detection of bus undervoltage within 10 seconds;
- b. All loads on each 480 V safeguards bus except for the safety related motor control centers, component cooling water (CCW) pump, and containment spray (CS) pump are capable of being tripped on an undervoltage signal (CCW pump must be capable of being tripped on coincident safety injection (SI) and undervoltage signal);
- c. The DG is capable of accepting required loads manually. Since most equipment which receives a SI signal are isolated in these MODES due to maintenance or low temperature overpressure protection concerns, and the DBA of concern (i.e., a fuel handling accident) would not generate a SI signal, manual loading of the DGs will most likely be required. These loads must be capable of being added to the OPERABLE DG within 10 minutes;
- d. The DG day tank is available to provide fuel oil for ≥ 1 hour at 110% design loads;
- e. The fuel oil transfer pump from the fuel oil storage tank to the associated day tank is OPERABLE including all required piping, valves, and instrumentation (long-term fuel oil supplies are addressed by LCO 3.8.3, "Diesel Fuel Oil"); and
- f. A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE.

(continued)

BASES

(continued)

BASES

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of postulated events and to maintain the plant in the cold shutdown or refueling condition are available.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, "AC Sources-MODES 1, 2, 3, and 4."

ACTIONS

A.1

As discussed in LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no onsite or offsite AC power to any required 480 V safeguards bus, the ACTIONS for LCO 3.8.10 must also be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite power circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a completely de-energized train.

(212)

With ~~no~~ offsite power available to one or more required 480 V safeguards bus(es) ~~inoperable~~, assurance must be provided that there is not a complete loss of required safety features. Although two trains may be required by LCO 3.8.10, one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, or operations involving positive reactivity additions. By allowing the option to declare required features inoperable that are not powered from offsite power, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

(continued)

BASES

ACTIONS
~~(continued)~~

~~A.2.1, A.2.2, and A.2.3~~

~~With the offsite power circuit not available to all required AC electrical trains, the option exists to declare all required features inoperable per Required Action and A.12.4~~

(continued)

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~~With the offsite power circuit not available to all required AC electrical trains, the option exists to declare all required features inoperable per Required Action A. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, and operations involving positive reactivity additions is an acceptable option to Required Action A. Performance of Required Actions A.1.2.1 and Performance of Required Actions A.2.2 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control. A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.~~

It is further required to immediately initiate action to restore the required offsite power AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required offsite power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

B.1, B.2, and B.3

(continued)

4 1 1



1 1 1

BASES

ACTIONS

212

B3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of temperature control within established procedures.

(continued)

D
BASES

1. ACTIONS

B.2.1, and B.3 (continued)

~~It is further required to immediately initiate action to restore the required DG to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary AC power redundancy to plant safety systems. B.3, and B.4 (continued)~~

212
It is further required to immediately initiate action to restore the required DG to OPERABLE status and to continue this action until restoration is accomplished in order to provide the necessary AC power redundancy to plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DG should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient redundant power.

L
SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

This SR requires the performance of SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6.

This SR precludes requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, precludes de-energizing a required 480 V safeguards bus, and precludes unnecessary transfers of the offsite power source configurations. With limited AC sources available, a single event could compromise both the required circuit and the DG. Therefore, the requirement to perform SR 3.8.1.3, and SR 3.8.1.6 through 3.8.1.9 is suspended. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil

BASES

BACKGROUND

*Change,
not in final
draft*

Fuel oil is provided to each emergency diesel generator (DG) by a dedicated 350 gal day tank located near the DG. Each day tank is supplied from an associated 6000 gal underground fuel oil storage tank. Each storage tank provides a minimum fuel oil capacity of 5000 gal. The two storage tanks are sufficient to operate both DGs at design ratings for 24 hours. The total minimum fuel oil capacity also ensures that both DGs can operate for a period of 40 hours while providing for a maximum post loss of coolant accident (LOCA) load demand. The maximum load demand is calculated using the assumption that both DGs are available and is less than the DG design rating. The minimum onsite fuel capacity is sufficient to operate the DGs for longer than 8 hours which is the time required to replenish the onsite supply from outside sources (Ref. 1).

Fuel oil is transferred from each storage tank to the associated day tank by a dedicated fuel oil transfer pump. Each fuel oil transfer pump is powered by a 480 V safeguards bus that is backed by the associated DG. One fuel oil transfer pump has the capability to supply both DGs operating with 110% of their design loads. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG.

All outside tanks, pumps, and piping are located underground to protect them from potential missiles. Heat tracing is provided in the exposed suction piping to the fuel oil pumps in the event that heating is lost in the DG rooms. The heat tracing is thermostatically controlled to maintain the fuel oil in the pipe > 40°F which is above the cloud point temperature of the fuel oil (0°F).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 2 and 3), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

Since diesel fuel oil supports the operation of the standby AC power sources, it satisfies Criterion 3 of the NRC Policy Statement.

LCO

Stored onsite diesel fuel oil is required to have sufficient supply for 40 hours of maximum post-LOCA load demand. It is also required to meet specific standards for quality. This requirement, in conjunction with an ability to obtain replacement fuel oil supplies within 8 hours, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4," and LCO 3.8.2, "AC Sources - MODES 5 and 6."

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil supports LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil is required to be within limits in MODES 1, 2, 3 and 4, and when the associated DG is required to be OPERABLE in MODES 5 and 6.

(continued)

BASES

ACTIONS

A.1

30

With one or more required DGs with an onsite supply of < 5000 gal of fuel oil, the assumed 40 hour fuel oil supply for a DG is not available. This circumstance may be caused by events, such as full load operation after an inadvertent start with an initial minimum required fuel oil level, or feed and bleed operations, which may be necessitated by increasing fuel oil particulate levels or any number of other oil quality degradations. Required Action A.1 allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. The Completion Time of 48 1/2 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity, the fact that actions will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

21

If one or more DGs has stored fuel oil with total particulates not within limits ~~for reasons not related to new fuel oil~~, the fuel oil must be restored within limits within 7 days. The fuel oil particulate properties of ~~viscosity, water, and sediment~~ are verified by SR 3.8.3.2. Trending of particulate levels normally allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample practices (bottom sampling), contaminated sampling equipment, or errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

(continued)

BASES

ACTIONS
(continued)

C.1

212

~~With a Required Action and associated Completion Time not met, or one or more DG's fuel oil properties defined in SR 3.8.3.2 not within required limits, a period of 30 days is allowed for reasons other than addressed by Conditions A or B (restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.~~

D.1

212

~~With a Required Action and associated Completion Time not met, or one or more DG's fuel oil not within limits for reasons other than addressed by Conditions A, B, or C (e.g., cloud point temperature reached), the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.~~

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.1

169

~~This SR verifies an onsite supply of ≥ 5000 gal of fuel oil is available for each required DG. This ensures that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 40 hours while providing maximum post-LOCA loads. The 40 hour period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.~~

each fuel oil storage tank contains

The Frequency of 31 days is adequate to ensure that a sufficient supply of fuel oil is available, since indications are available to ensure that operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

(continued)

BASES

SURVEILLANCE

SR 3.8.3.2

(continued)

REQUIREMENTS

This SR provides a means of determining whether new and stored fuel oil has been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. This ensures the availability of high quality fuel oil for the DGs. Fuel oil degradation during long term storage is indicated by an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which could eventually cause engine failure.

A fuel oil sample is analyzed to establish that properties specified in Table 1 of ASTM D975-78 (Ref. 4) for viscosity, water, and sediment are met for the stored fuel oil.

~~SURVEILLANCE — SR 3.8.3.2 (continued)~~

REQUIREMENTS

The Frequency of this SR takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals. The Frequency, as specified in the Diesel Fuel Oil Testing Program, is 92 days.

REFERENCES

1. UFSAR, Section 9.5.4.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. ASTM Standards, D975-78, Table 1.
-

AC Sources - MODES 1, 2, 3, and 4
B 3.8.1

DC Sources - MODES 1, 2, 3, and 4
B 3.8.4

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential active components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for these two electrical power sources which are further divided and organized based on voltage considerations and whether they are Class 1E (i.e., supply safety related or engineered safeguards functions) or nonessential. This LCO is provided to specify the minimum sources of DC power which are required to supply the DC buses and their associated distribution system during MODES 1, 2, 3, and 4.

The station DC electrical power subsystem provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC instrument bus power (via inverters). Atomic Industrial Forum (AIF) GDC 39 (Ref. 1) requires emergency power sources be provided and designed with adequate independence, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems.

The 125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power distribution train (Train A and Train B). Each subsystem consists of one 125 VDC battery, two battery chargers supplied from the 480 V system, distribution panels and buses, and all the associated control equipment and interconnecting cabling (see Figure B 3.8.4-1). The batteries and battery chargers are addressed by this LCO.

(continued)

BASES

(continued)

BASES

BACKGROUND
(continued)

Each battery provides a separate source of DC power independent of AC power. Each of the two batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 105 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 2).

There are four battery chargers available to the batteries. Chargers 1A and 1B are rated at 150 amps and chargers 1A1 and 1B1 are rated at 200 amps. Battery chargers 1A and 1A1 are normally aligned to battery A, and battery chargers 1B and 1B1 are normally aligned to battery B. A charging capacity of at least 150 amps is normally required to supply the necessary DC loads on each train and to provide a full battery charge to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated Design Basis Accident (DBA). The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System—MODES 1, 2, 3, and 4," and LCO 3.8.10, "Distribution Systems—MODES 5 and 6."

The DC electrical power distribution subsystem also provide DC electrical power to the inverters, which in turn power the AC instrument buses. The inverters are described in more detail in Bases for LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4," and LCO 3.8.8, "AC Instrument Bus Sources—MODES 5 and 6."

Train A Engineered Safety Feature (ESF) equipment is supplied from battery A, while Train B ESF equipment is supplied from battery B. Additionally, the 480 V ESF switchgear and diesel generator (DG) control panels are supplied from either battery by means of an automatic transfer circuit in the switchgear and control panels. The normal supply from Train A (Buses 14 and 18 and DG A) is from DC distribution panels A. These panels also provide the emergency DC supply for Train B. Similarly, the normal supply from Train B (Buses 16 and 17 and DG B) is from DC distribution panels B. These panels also provide the emergency dc supply for Train A.

(continued)

BASES

BACKGROUND
(continued)

Each 125 VDC battery and associated battery chargers are separately housed in a ventilated room with its associated distribution center. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. The two battery rooms are supplied with ventilation by a common AC powered air conditioning and heating unit which also provides sufficient air changes to prevent hydrogen buildup. A redundant DC powered fan is also available in the event that all AC power is lost. The failure of both the AC powered and DC powered units does not result in unacceptable room service conditions until after 5 hours of continuous battery operation during a DBA (Ref. 2).

The batteries for Train A and Train B DC electrical power distribution subsystem are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity for aging considerations. The minimum voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery.

Each battery charger for the Train A and Train B DC electrical power distribution subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 2).

APPLICABLE
SAFETY ANALYSES

(149)

The initial conditions of a DBA and transient analyses (Refs. 3, 4, and 4-5), assume that ESF systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one train of DC source OPERABLE in the event of:

- a. An assumed loss of all offsite AC power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the DC electrical power sources ensures that one train of DC electrical power is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one DC electrical power source).

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of DC power ensures that at least one DC power source is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 6). Therefore, the requirements of GDC 17 (Ref. 7) can be met at all times.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

(continued)



BASES (continued)

LCO

The Train A and Train B DC electrical power sources, each consisting of one battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA. Loss of any one train DC electrical power source does not prevent the minimum safety function from being performed.

(169) An OPERABLE DC electrical power source requires the battery and at least one battery ~~battery~~-charger with a capacity \geq 150 amps to be operating and connected to the associated DC bus. The AC powered and DC powered fan units are not required to be OPERABLE for this LCO, but some form of ventilation may be required for SR 3.8.6⁴² and SR 3.8.6.5

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe plant operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in LCO 3.8.5, "DC Sources—MODES 5 and 6."

(continued)

BASES (continued)

ACTIONS

A.1

With one DC electrical power source inoperable, OPERABILITY must be restored within 2 hours. In this Condition, redundancy is lost and only one train is capable to completely respond to an event. If one of the required DC electrical power sources is inoperable, the remaining DC electrical power source has the capacity to support a safe shutdown and to mitigate an accident condition. A subsequent worst case single failure would, however, result in the complete loss of the remaining 125 VDC electrical power distribution subsystem with attendant loss of ESF functions. The 2 hour Completion Time reflects a reasonable time to assess plant status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power source is not restored to OPERABLE status, to prepare to effect an orderly and safe plant shutdown.

B.1 and B.2

(129)

If the inoperable DC electrical power source cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both DC electrical power sources are inoperable, a loss of multiple safety functions exists; therefore, LCO 3.0.3 must be immediately entered.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.4.1

234

~~This SR verifies that each battery has a battery charger with a capacity of ≥ 150 amps. Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. This verification helps to ensure float charge is the effectiveness of the charging system condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and the ability of the batteries to perform their intended function maintain the battery (or a battery cell) in a fully charged state. The frequency of 31 days is considered acceptable voltage requirements are based on operating experience the nominal design voltage of the battery and other indications available are consistent with the initial voltages assumed in the control room that alert the operator to battery malfunctions battery sizing calculations. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).~~

SR 3.8.4.2

This SR verifies that the capacity of each battery is adequate to supply and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test. A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements specified in Reference 2.

149

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 5-9) and Regulatory Guide 1.129 (Ref. 6-10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 24 months.

This SR is modified by two Notes. Note 1 states that SR 3.8.4.3 may be performed in lieu of SR 3.8.4.2. This substitution is acceptable because SR 3.8.4.3 represents a

(continued)

BASES (continued)

more severe test of battery capacity than does SR 3.8.4.2. Note 2 states that this surveillance shall not be performed in MODE 1, 2, 3, or 4 because performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

(continued)

BASES (continued)

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.4.3

This Surveillance verifies that each battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test. A battery performance test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity as determined by specified acceptance criteria. The test is intended to determine overall battery degradation due to age and usage.

A battery should be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Frequency for this SR is 60 months when the battery is $< 85\%$ of its expected life with no degradation and 12 months if the battery shows degradation or has reached 85% of its expected life with a capacity $< 100\%$ of the manufacturer's rating. When the battery has reached 85% of its expected life with capacity $\geq 100\%$ of the manufacturer's rating, the Frequency becomes 24 months. Battery degradation is indicated when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is $\geq 10\%$ below the manufacturer rating. These Frequencies are considered acceptable based on the testing being performed in a conservative manner relative to the battery life and degradation. This ensures that battery capacity is adequately monitored and that the battery remains capable of performing its intended function.

This SR is modified by a Note stating that this SR shall not be performed in MODE 1, 2, 3, or 4. The reason for the Note is that during operation in these MODES, performance of this SR could cause perturbations to the electrical distribution system and challenge safety systems.

(continued)

BASES (continued)

- REFERENCES
1. Atomic Industrial Forum (AIF) GDC 39, Issued for comment July 10, 1967.
 2. UFSAR, Section 8.3.2.
 3. UFSAR, Section 9.4.9.3.
 4. UFSAR, Chapter 6.
 5. UFSAR, Chapter 15.
 6. UFSAR, Section 8.3.1.
 7. 10 CFR 50, Appendix A, GDC 17.
 8. ~~IEEE-450-1980~~

~~THE FOLLOWING TEXT WAS MOVED~~

9.

~~THE PRECEDING TEXT WAS MOVED~~

Regulatory Guide 1.32, February 1977.

~~10~~ Regulatory Guide 1.129, December 1974.

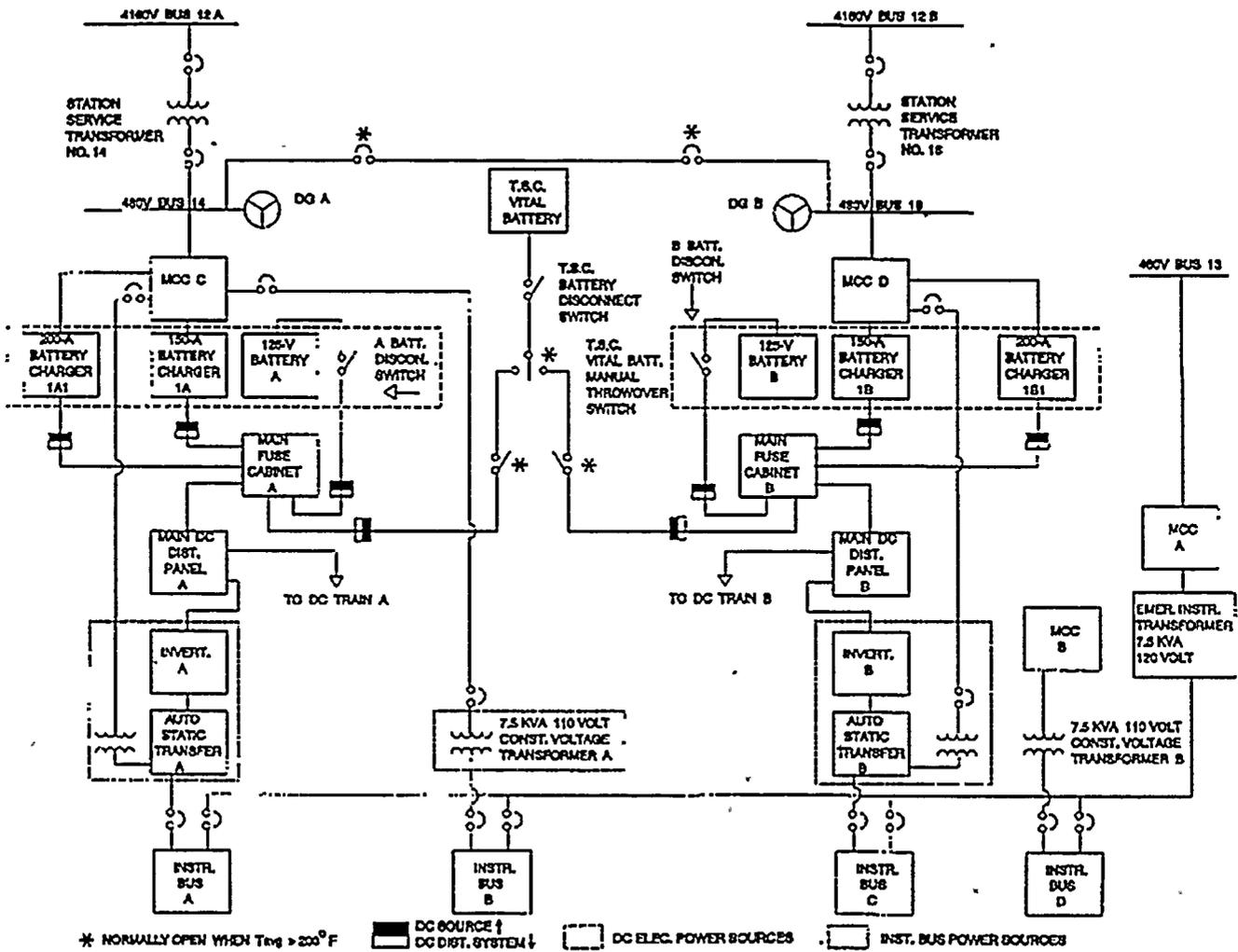


Figure B 3.8.4-1

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.5, "DC Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

(169) In MODE 5 or 6, the number of required DC electrical sources may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number ~~of~~ required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the DC electrical sources, must be removed from service. The minimum required DC electrical sources is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 ensures that:

- a. Required features needed to mitigate a fuel handling accident are available;
- b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the DC electrical power sources ensures that one train of DC sources are OPERABLE in the event of:

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

(149)

The DC electrical power sources are required to be OPERABLE to support the distribution subsystems required OPERABLE by LCO 3.8.10, "~~Distribution Systems - Shutdown Systems - MODES 5 and 6.~~" If only one DC electrical power distribution train is required to be OPERABLE, the minimum source consists of a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the required train. If both DC electrical power trains are required, one DC source must contain a battery, a charging capacity of at least 150 amps, and the corresponding control equipment and interconnecting cabling within the train system. The second DC source may consist of only a battery charger with a capacity of at least 150 amps, or a battery, and the corresponding control equipment and interconnecting cabling. The two must be sufficiently independent that a loss of all offsite power sources, a loss of onsite standby power, or a worst case single failure does not affect more than one required DC electrical power train. This ensures the availability of sufficient DC electrical power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The AC powered and DC powered fan ventilation units are not required to be OPERABLE for this LCO, but some form of ventilation may be required to meet SR 3.8.6⁴³²

and SR 3.8.6.5

(149)

(continued)

BASES

APPLICABILITY The DC electrical power sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the affects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4."

ACTIONS

A.1

Although two trains may be required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," the remaining DC electrical train may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS, and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable associated with the required inoperable DC power source(s), appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. Required features remaining powered from a DC electrical source, even if that source is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

(continued)

D
BASES

ACTIONS
(continued)

A.2.1, A.2.2, A.2.3, and A.2.4

212
With one or more required DC electrical power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

It is further required to immediately initiate action to restore the required DC electrical power source and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE
REQUIREMENTS

SR 3.8.5.1

149
~~SR 3.8.5.1~~ This SR requires the performance of SRs from LCO 3.8.4 that are necessary for ensuring the OPERABILITY of the DC electrical power subsystem in MODES 5 and 6.

(continued)



BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.5.1 (continued)

This SR precludes requiring the OPERABLE DC electrical power source from being removed from service to perform a battery service test or a performance discharge test. With limited DC sources available, a single event could compromise multiple required safety features. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DC electrical power source is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.4 for a discussion of the specified SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

Each DC electrical power train contains a 125 VDC battery which is capable of carrying the expected shutdown loads following a plant trip and a loss of all AC power for a period of 4 hours without battery terminal voltage falling below 105 V. Major loads and approximate operating times on each battery are discussed in the UFSAR (Ref. 1). The batteries are normally in standby since the associated battery chargers provide for the required DC system loads.

The batteries for Train A and Train B DC electrical power are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and 100% design demand. Battery size is based on 125% of required capacity for aging considerations.

(212)

~~The minimum voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery.~~

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries to ensure that the batteries are capable of performing their safety function as required by LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4," and LCO 3.8.5, "DC Sources - MODES 5 and 6."

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

(109)

The initial conditions of Design Basis Accident (DBA) and transient analyses assume Engineered Safety Feature systems are OPERABLE (Refs. 2 and 3). The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation. The DC sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to Engineered Safety Feature systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

Battery cell parameters satisfy Criterion 3 of the NRC Policy Statement.

LCO

(22)

This LCO requires that battery cell parameters for Train A and B batteries be within limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery cell parameters are defined for electrolyte level, temperature, float voltage, and specific gravity. The limits for electrolyte level, float voltage, and specific gravity are conservatively established for both designated pilot cells and connected cells (Category A and B, respectively of Table B-3.8.6-1) within plant procedures. Failure to meet these established limits may allow continued DC electrical system function even with Category A and B limits not met for a limited duration provided that the upper limit specified in the associated Surveillance Requirement for each connected cell (Category C) is not exceeded. In addition, the average electrolyte temperature of the term "connected cell" excludes any battery cells that may be $\geq 65^{\circ}\text{F}$ jumpered out.

(continued)

BASES (continued)

212 ~~The battery cell parameters are specified in Table B-3.8.6-1. This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.~~

(continued)

BASES (continued)

~~LCO~~ ~~Category A~~
~~(continued)~~

~~Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, level, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.~~

~~The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE 450 (Ref. 3), with an additional allowance of 1/4 inch above the high water level indication mark for operating margin to account for temperature and charge effects. In addition to this allowance, footnote a to Table B 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. The specified maximum level is defined as 1/4 inch above the maximum indication mark. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE 450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.~~

~~The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendations of IEEE 450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.~~

~~The Category A limit specified for specific gravity for each pilot cell is ≥ 1.193 for Battery A and ≥ 1.197 for Battery B (0.015 below the manufacturer fully charged, nominal specific gravity) or a battery charging current that is stabilized at a value of < 2 amps. This value is characteristic of a charged cell with adequate capacity.~~

~~According to IEEE 450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C). The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.~~

(continued)

BASES (continued)

~~LCO~~ ~~Category A (continued)~~

~~Footnote b to Table B 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition and can be used as an alternative to specific gravity.~~

Category B

~~Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.~~

~~The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B (0.020 below the manufacturer fully charged, nominal specific gravity) or a battery charging current that is stabilized at a value < 2 amps. The average of all connected cells must also be > 1.198 for Battery A and > 1.202 for Battery B (0.010 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that for Category A and has been discussed above.~~

(continued)

BASES (continued)

~~LCO~~ ~~Category C~~
~~(continued)~~

~~Category C defines the limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out. These limits, although reduced from the Category A and B limits, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limit, the assurance of sufficient capacity described above no longer exists, and the battery must immediately be declared inoperable.~~

~~The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE 450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.~~

~~The Category C limit for the average of all connected cells specifies a specific gravity ≥ 1.188 for Battery A and ≥ 1.192 for Battery B (0.020 below the manufacturer recommended fully charged, nominal specific gravity). These values are based on manufacturer recommendations. In addition to that limit, it is required that the specific gravity for each connected cell must be no more than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that for Category A and has been discussed above.~~

~~Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE 450 (Ref. 3).~~

(continued)

BASES (continued)

APPLICABILITY

(212)

The battery cell parameters for Train A and Train B batteries are required solely for the support of the associated DC electrical power subsystem. Therefore, the battery cell parameter limits ~~of Table B-3.8.6-1~~ are required to be met when the DC power source is required to be OPERABLE. Since the Train A and Train B batteries support LCO 3.8.4 and LCO 3.8.5, the battery cell parameters ~~of Table B-3.8.6-1~~ are required to be met in MODES 1, 2, 3, and 4, and when the associated DC electrical power subsystems are required to be OPERABLE in MODES 5 and 6.

ACTIONS

(212)

The ACTIONS are modified by a Note to provide clarification that separate condition entry is allowed for each battery. Separate Condition entry is acceptable since the battery cell parameters ~~of Table B-3.8.6-1~~ are provided on a battery basis.

A.1

(212)

With one or more batteries with one or more battery cell parameters outside the limits for any connected cell, Asufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power train must be immediately declared inoperable and actions taken per LCO 3.8.4 or LCO 3.8.5.

(continued)

BASES (continued)

~~2. SURVEILLANCE~~ SR 3.8.6.1
~~REQUIREMENTS~~

~~This SR verifies that the electrolyte level of each connected battery cell is above the top of the plates and Anot overflowing. 3~~

212
~~With one or more batteries this is consistent with one or more battery cell parameters not within limits (IEEE-450 (Ref. e4) and ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. Category A limits not met, Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table B 3.8.6-1 the battery cells are degraded but the battery still has sufficient capacity to perform its intended function. The Frequency of 31 days is consistent with IEEE-450.—~~

~~Therefore, the affected battery is not required to be considered inoperable per LCO 3.8.4 or LCO 3.8.5 solely as a result SR: 3.8.6.2~~

~~This SR verifies that the float voltage of Category A or B limits not met and operation is permitted for a limited period before the each connected battery cell parameters must be restored within limits is > 2.07 V.~~

~~With one or more batteries with one or more battery cell parameters not within Category A or B limits, the pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits of Table B 3.8.6-1 within 1 hour. This check will provide a quick indication of the status of the remainder of the battery cells. The Completion Time of one hour provides sufficient time to inspect the electrolyte level and to confirm the float voltage of the pilot cells.—~~

(continued)

BASES (continued)

~~ACTIONS~~ ~~A.1, A.2, and A.3~~ (continued)

~~If the pilot cell's electrolyte level and float voltage are found to be within the Table B 3.8.6-1 Category C limits, all connected cells must be verified to be within Category C limits within 24 hours. Completion Time of 24 hours is allowed to complete the initial verification. This takes into consideration both the time required to perform the required verification (including specific gravity measurements) and the assurance that the battery pilot cell parameters are not severely degraded. This verification is required every 7 days until all battery cell parameters are restored to Category A and B limits. This periodic verification is more restrictive than the normal frequency of pilot cell surveillances.~~

~~Battery cell parameters must be restored to within Category A and B limits within 31 days. A Completion Time of 31 days to restore battery cell parameters to within limits is acceptable since the battery remains capable of performing its intended function in this condition.~~

B.1

~~If the Required Action and associated Completion Time of Condition A are not met, or with one or more batteries with an average electrolyte temperature of representative cells $< 65^{\circ}\text{F}$, or with one or more batteries with one or more battery cell parameters outside the Category C limits for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power train must be immediately declared inoperable and actions taken per LCO 3.8.4 or LCO 3.8.5.~~

(continued)



BASES (continued)

~~SURVEILLANCE~~ ~~SR 3.8.6.1~~
REQUIREMENTS

(212) This SR verifies that Table B-3.8.6-1 Category A battery cell parameters are consistent with IEEE 450 (Ref. 3), which recommends regular battery inspections including float voltage, specific gravity, and electrolyte level of each pilot cell. The Completion Time of one hour provides sufficient time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. This limit is based on IEEE-450 (Ref. —

(continued)

BASES (continued)

~~ACTIONS~~ A4) which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement. 1—The frequency of 31 days is also consistent with IEEE-450.

~~2, and SR 3.8.6.3~~

This SR verifies the specific gravity of the designated pilot cell in each battery is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B. 3—(continued)

(212) If the pilot cell's electrolyte level and float voltage these values are found to be within the Table B 3.8.6-1 Category C limits, all connected cells must be verified to be within Category C limits within 24 hours based on manufacturer recommendations. Completion Time of 24 hours is allowed according to complete the initial verification IEEE 450 (Ref.— This takes into consideration both the time required to perform the required verification (including 4) the specific gravity measurements) and the assurance that the battery pilot cell parameters readings are not severely degraded based on a temperature of 77°F (25°C). This verification is required every 7 days until all battery cell parameters the specific gravity readings are restored to Category A corrected for actual electrolyte temperature and B limits level. This periodic verification is more restrictive than for each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the normal frequency of pilot cell surveillance reading; 1 point is subtracted for each 3°F below 77°F.

Battery The specific gravity of the electrolyte in a cell parameters must be restored increases with a loss of water due to within Category A and B limits within 31 days electrolysis or evaporation.—

A Completion Time of 31 Because of specific gravity gradients that are produced during the recharging process, delays of several days to restore battery cell parameters to within limits is acceptable since the battery remains capable of performing its intended function in this condition may occur while waiting for the specific gravity to stabilize.

(continued)

BASES (continued)

BA stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. †

212
~~If the Required Action and associated Completion Time of Condition A are not met, or with one or more batteries with an average electrolyte temperature of representative cells < 65°F, or with one or more batteries with one or more battery cell parameters outside the Category C limits for any connected cell, sufficient capacity to supply the maximum expected load requirement. This phenomenon is not assured and the corresponding DC electrical power train must be immediately declared inoperable and actions taken per LCO 3.8.4 or LCO 3.8.5 further discussed in IEEE-450.~~

(continued)

BASES (continued)

SURVEILLANCE — SR 3.8.6.1
REQUIREMENTS

This SR verifies that ~~Table B 3.8.6.1 Category A battery cell parameters are~~ The Frequency of 31 days is consistent with IEEE 450 (Ref IEEE 450).

(212)

(continued)

BASES (continued)

~~3), which recommends regular battery inspections including float voltage, specific gravity, and~~ SURVEILLANCE SR 3.8.6.4

REQUIREMENTS
(continued)

This SR verifies the average electrolyte level temperature of each the designated pilot cell in each battery is $\geq 55^{\circ}\text{F}$. This temperature limit is an initial assumption of the battery capacity calculations. The Frequency of 31 days is consistent with IEEE-450 (Ref. 3)4).

SR 3.8.6.23.8.6.5

212 This SR verifies that Table B 3.8.6-1 Category B battery the average temperature of every fifth cell parameters are consistent with IEEE-450 (Ref. 3) of each battery is $\geq 55^{\circ}\text{F}$. 3). The battery inspection shall include float voltage, specific gravity, and electrolyte level of each connected cell. The Frequency of 92 days is consistent with IEEE 450 (Ref. 3). In addition, within 7 days of a battery discharge $< 105\text{ V}$ or a battery overcharge $> 150\text{ V}$, the battery must be demonstrated to meet Category B limits. This inspection is also consistent with IEEE 450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This SR verifies that the average temperature of representative cells is $\geq 65^{\circ}\text{F}$. This is consistent with the recommendations of IEEE-450 (Ref. 3)4). Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. The Frequency of 92 days is consistent with IEEE-450 (Ref.-

3)SR 3.8.6.6

This SR verifies the specific gravity of each connected cell is not more than 0.020 below average of all connected cells and that the average of all connected cells is ≥ 1.188 for Battery A and ≥ 1.192 for Battery B.

BASES (continued)

REFERENCES

1. These values are based on manufacturer recommendations and IEEE-450 (Ref. 4) which ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery. The temperature correction for specific gravity readings is the same as that discussed for SR 3.8.6.3. The frequency of 92 days is consistent with IEEE-450.

REFERENCES

1. UFSAR, Section 3.8.2.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. IEEE-450-1980.
-
-

BASES (continued)

Table B-3.8.6-1 (page
AC Instrument Bus Sources - MODES 1, 2, 3, and 4
B 3.8.7

~~B 3.8 ELECTRICAL POWER SYSTEMS~~

~~B 3.8.7 AC Instrument Bus Sources - MODES 1, 2, 3, and 4~~

~~BASES~~

~~BACKGROUND~~

~~The AC instrument bus electrical power distribution subsystem consists of 1) Battery Cell Parameters Requirements~~

~~Four 120 VAC instrument buses~~

- (a) ~~It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.~~
- (b) ~~Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.~~

~~B-3.8 ELECTRICAL POWER SYSTEMS~~

~~B-3.8.7 AC Instrument Bus Source MODES 1, 2, 3, and 4~~

~~BASES~~

~~BACKGROUND~~ The AC instrument bus electrical power distribution subsystem consists of four 120 VAC instrument buses. Not more than 0.020 below average of all connected cells

~~AND~~

~~Average of all connected cells:~~

~~≥ 1.188 for Battery A and ≥ 1.192 for Battery B
 ≥ 1.188 for Battery A and ≥ 1.192 for Battery B~~

~~AND~~

~~Average of all connected cells:~~

~~> 1.198 for Battery A and > 1.202 for Battery B~~

~~≥ 1.193 for Battery A and ≥ 1.197 for Battery B
Specific Gravity(b)~~

~~> 2.07 V~~

~~≥ 2.13 V~~

~~≥ 2.13 V~~

~~Float Voltage~~

~~Above top of plates, and not overflowing~~

~~$>$ minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark(a)~~

~~$>$ minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark(a)~~

~~Electrolyte Level(a) CATEGORY C: ALLOWABLE LIMIT FOR EACH CONNECTED CELL
CATEGORY B: LIMITS FOR EACH CONNECTED CELL CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL~~

PARAMETER The power source for one 120 VAC instrument bus (Instrument Bus D) is normally supplied from offsite power via a non-Class 1E constant voltage transformer (CVT) such that only three buses are considered safety related (see

(continued)



BASES (continued)

Figure 3.8.4-1). These three 120 VAC instrument buses (A, B, and C) supply a source of power to instrumentation and controls which are used to monitor and actuate the Reactor Protection System (RPS) and Engineered Safety Features (ESF) and other components (Ref. 1). The loss of Instrument Bus D is addressed in LCO 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and LCO 3.3.3, "Post-Accident Monitoring Instrumentation."

Instrument Buses A and C can be supplied power either from inverters which are powered from separate and redundant DC power sources, a non-Class 1E CVT (maintenance CVT) powered from offsite power, or a Class 1E CVT (see Figure B 3.8.4-1). The inverters are the preferred source of power for Instrument Bus A and C because of the stability and reliability they achieve.

Instrument Bus B can be supplied power from either a Class 1E CVT or a non-Class 1E CVT (maintenance CVT) powered from offsite power. The Class 1E CVT, supplied by motor control center C (MCC C is supplied by 480 V safeguards Bus 14), is the preferred source of power for Instrument Bus B because of the potential to have a power interruption if offsite power were unavailable.

(continued)

BASES (continued)

BACKGROUND
(continued)

The majority of instrumentation and controls supplied by the 120 VAC instrument buses are fail safe devices such that they go to their post accident position upon loss of power. However, a notable exception to this is the actuation logic for Containment Spray (CS) System which requires 120 VAC and 125 VDC power in order to function. This prevents a spurious CS actuation from occurring if control power were lost. The actuation logic for CS is powered from all three instrument buses and from both DC electrical power distribution trains.

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 2 and 3), assume Engineered Safety Feature systems are OPERABLE. The AC instrument bus power sources are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESF instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC instrument bus power sources is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the plant. This includes maintaining required AC instrument buses OPERABLE in the event of:

- a. An assumed loss of all offsite AC electrical power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC instrument bus power sources ensures that one train of AC instrument buses are available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one AC instrument bus power source).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of AC instrument bus power also ensures that at least one train of AC instrument buses is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater pump during the estimated 8 hours required to provide this power transfer (Ref. 4). Therefore, the requirements of GDC 17 (Ref. 5) can be met at all times.

The AC instrument bus sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

The AC instrument bus sources ensure the availability of 120 VAC electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESF instrumentation and controls is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source for the third instrument bus.

(continued)

BASES (continued)

LCO
(continued)

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4").

For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4").

APPLICABILITY

The AC instrument bus power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

AC instrument bus power requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "~~Inverters - MODES~~" AC Instrument Bus Sources - MODES 5 and 6."

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ACTIONS

A.1, A.2, and A.3

With an inverter inoperable, its associated AC instrument bus becomes inoperable until it is re-energized from either its Class 1E or non-Class 1E CVT.

Required Action A.1 allows the instrument bus to be powered from either its associated Class 1E CVT or from a non-Class 1E CVT. For Instrument Buses A and C, the non-Class 1E power is supplied by a non-safety related motor control center (MCC A) which is supplied by 480 V Bus 13. The Completion Time of 2 hours is consistent with LCO 3.8.9, "Distribution Systems - MODES 1, 2, 3, and 4".

(continued)

BASES (continued)

ACTIONS

A.1, A.2, and A.3 (continued)

Required Action A.2 is intended to limit the amount of time that the instrument bus can be connected to a non-Class 1E power supply. The 24 hour Completion Time is based upon engineering judgement, taking into consideration the time required to repair the Class 1E CVT or the inverter and the additional risk to which the plant is exposed because of the connection to a non-Class 1E power supply.

Required Action A.3 allows 72 hours to fix the inoperable inverter and restore it to OPERABLE status. The 72 hour Completion Time is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC instrument bus is powered from its CVT, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible, battery backed inverter source to the AC instrument buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

With the Class 1E CVT for Instrument Bus B inoperable, the instrument bus becomes inoperable until it is re-energized from its non-Class 1E CVT. Required Action B.1 requires Instrument Bus B to be powered from its non-Class 1E CVT within 2 hours. The non-Class 1E power is supplied by a nonsafety related 480 V motor control center (MCC A) which is supplied by 480 V Bus 13.

(continued)

BASES (continued)

ACTIONS

B.1 and B.2 (continued)

Required Action B.2 is intended to limit the amount of time that Instrument Bus B can be connected to a non-Class 1E power supply. The 7 day limit is based on engineering judgement, taking into consideration the time required to repair the Class 1E CVT and the additional risk to which the plant is exposed because of the Class 1E CVT inoperability. This must be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When Instrument Bus B is powered from its non-Class 1E CVT, it is relying upon interruptible offsite AC electrical power sources. The Class 1E, diesel generator backed, CVT to Instrument Bus B is the preferred power source for powering instrumentation trip setpoint devices.

C.1 and C.2

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If the inoperable devices or components cannot be restored to OPERABLE status or other Required Actions are not completed within the required Completion Time of Condition A or B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If two or more required AC instrument bus power sources are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when both inverters, or one or more inverters and the Class 1E CVT to Instrument Bus B are discovered to be inoperable.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.7.1

This SR verifies correct static switch alignment to Instrument Bus A and C. This verifies that the inverters are functioning properly and AC Instrument Bus A and C are energized from their respective inverter. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument buses. The Frequency of 7 days takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

SR 3.8.7.2

This SR verifies the correct Class 1E CVT alignment to Instrument Bus B. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

1. UFSAR, Chapter 8.3.2.
 2. UFSAR, Chapter 6.
 3. UFSAR, Chapter 15.
 4. UFSAR, Section 8.3.1.
 5. 10 CFR 50, Appendix A, GDC 17.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 AC Instrument Bus Sources—MODES 5 and 6

BASES

BACKGROUND.

The Background section of the Bases for LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6, the number of required AC instrument buses may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the AC instrument bus sources, must be removed from service. The minimum required AC instrument bus electrical subsystem is based on the requirements of LCO 3.8.10, "Distribution Systems—MODES 5 and 6."

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum AC instrument bus power sources to each required AC instrument bus during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC instrument bus power sources ensures that one train of the AC instrument buses are OPERABLE in the event of:

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC instrument bus power sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESF instrumentation and controls is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source for the third instrument bus.

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.4, "DC Sources - MODES 1, 2, 3, and 4).

(continued)



BASES

LCO
(continued)

For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4). Power sources ensure the availability of sufficient power to the required AC instrument buses to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available.

AC Instrument Bus power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1

Although two trains may be required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," the remaining OPERABLE AC instrument bus train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable with the associated AC instrument bus power source inoperable, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. This condition must be entered when the inverters for Instrument Bus A or C are inoperable, or the Class 1E CVT for Instrument Bus B is inoperable.

(continued)

BASES

ACTIONS

(continued)

A.2.1, A.2.2, A.2.3, and A.2.4

~~With one or more required AC instrument bus power sources inoperable, and the option exists to declare all required features inoperable per Required Action A.2-5~~

(continued)

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movement of

~~With one or more required AC instrument bus power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions. Therefore, immediate suspension of CORE ALTERATIONS, irradiated fuel assemblies, and operations involving positive reactivity additions is made an acceptable option to Required Action A. Therefore, immediate suspension of CORE ALTERATIONS and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.~~

It is further required to immediately initiate action to restore the required AC instrument bus power source and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC instrument bus power source should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from an alternate power source.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

(164)

This SR verifies correct static switch alignment to the required AC instrument buses. This SR verifies that the inverter is functioning properly and the AC instrument bus is energized from the inverter. The verification ensures

(continued)

BASES

that the required power is available for the instrumentation connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant capability of the inverter and other indications available in the control room that alert the operator to inverter malfunctions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.8.2

This SR verifies the correct Class 1E CVT alignment when Instrument Bus B is required. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. 3 The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

None.



B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - MODES 1, 2, 3, and 4

BASES

BACKGROUND

A source of electrical power is required for most safety related and nonessential action components. Two sources of electrical power are available, alternating current (AC) and direct current (DC). Separate distribution systems are developed for each of these electrical power sources which are further divided and organized based on voltage considerations and safety classification. This LCO is provided to specify the AC, DC, and AC instrument bus power electrical power distribution subsystems which are required to supply safety related and Engineered Safety Feature (ESF) Systems in MODES 1, 2, 3, and 4.

The onsite Class 1E AC, DC, and AC instrument bus electrical power distribution subsystems are each divided into two redundant and independent distribution trains. Each of these electrical power distribution subsystems, and their trains, are discussed in detail below.

(continued)

BASES (continued)

BACKGROUND
(continued)

AC Electrical Power Distribution Subsystem

The Class 1E AC electrical power distribution subsystem is organized into two redundant and independent trains (Train A and Train B). Each train consists of two 480 V safeguards buses, distribution panels, motor control centers and load centers (see Figure B 3.8.1-1). The 480 V safeguards buses for each train are capable of being supplied from two sources of offsite power as well as a dedicated onsite emergency diesel generator (DG) source. These power sources are discussed in more detail in the Bases for LCO 3.8.1, "AC Sources—MODES 1, 2, 3, and 4." The 480 V safeguards buses in turn supply motor control centers, distribution panels and load centers which supply motive power to required motor operated valves, pumps, dampers, or any other component which requires AC power to perform its safety related function. The AC electrical power distribution subsystem also supplies one of the three required AC instrument buses through a constant voltage transformer and provides a backup source for the other two instrument buses. The list of all required AC 480 V safeguards buses is provided in Table B 3.8.9-1.

BACKGROUND——DC Electrical Power Distribution Subsystem

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—(continued)

The Class 1E DC electrical power distribution subsystem is organized into two redundant and independent trains (Train A and Train B). Each train consists of a Class 1E battery and two battery chargers (with a charging capacity of at least 150 amps) which supply a main 125 VDC distribution panel (see Figure B 3.8.4-1). These power sources are discussed in more detail in the Bases for LCO 3.8.4, "DC Sources—MODES 1, 2, 3, and 4." Each main distribution panel supplies secondary distribution panels which provide control power to AC powered components and control power for other devices such as solenoid operated valves and air operated valves. The DC electrical power distribution subsystem also supplies two of the four AC instrument buses through inverters. The list of all required DC distribution panels is provided in Table B 3.8.9-1.

(continued)

BASES (continued)

(continued)

BASES (continued)

BACKGROUND
(continued)

AC Instrument Bus Electrical Power Distribution Subsystem

The AC instrument bus electrical power distribution subsystem consists of four 120 VAC instrument buses. The power source for one 120 VAC instrument bus (Instrument Bus D) is supplied from offsite power via a non Class 1E constant voltage transformer (CVT) such that only three buses are considered safety related (see Figure ~~B3.8.4-1~~ ~~3.8.4-1~~). These three buses are organized into two redundant and independent trains (Train A and Train B). These trains supply a source of power to instrumentation and controls which are used to monitor and actuate ESF and other components. Train A consists of two buses with one bus (Instrument Bus A) normally powered from an inverter and the other (Instrument Bus B) normally powered from a Class 1E CVT. Train B consists of one bus (Instrument Bus C) normally powered from an inverter. The long-term alternate power supplies for Instrument Bus A and C are two Class 1E CVTs, each powered from the same train as the associated battery chargers, and their use is governed by LCO 3.8.7, "AC Instrument Bus Sources—MODES 1, 2, 3, and 4." The list of required 120 VAC instrument buses is provided in Table B 3.8.9-1. The loss of Instrument Bus D is addressed in LCO 3.3.2, "Engineered Safety Feature Actuation Actuation System (ESFAS) Instrumentation," and LCO 3.3.3, "Post-Accident Monitoring (PAM) Instrumentation."

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(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses (Refs. 1 and 2) assume ESF systems are OPERABLE. The AC, DC, and AC instrument bus electrical power distribution subsystems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, "Power Distribution Limits;" Section 3.4, "Reactor Coolant System (RCS);" and Section 3.6, "Containment Systems."

The OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining power distribution subsystems OPERABLE in the event of:

- a. An assumed loss of all AC offsite power or all onsite standby AC power; and
- b. A worst case single failure.

In the event of a DBA, the OPERABILITY requirements of the AC, DC, and AC instrument bus electrical power distribution subsystems ensures that one train of each distribution subsystem is available with:

- a. An assumed loss of all offsite power; and
- b. A worst case single failure (including the loss of one train of offsite standby AC power).

(continued)



BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

In general, the accident analyses assume that all offsite power is lost at the time of the initiating event except where the availability of offsite power provides worst case conditions (e.g., steam line break with continued operation of the reactor coolant pumps). The availability of redundant offsite power sources (i.e., circuits 751 and 767) helps to reduce the potential to lose all offsite power. Providing redundant sources of offsite power also ensures that at least one AC, DC, and AC instrument bus train is available if all onsite standby AC power is unavailable coincident with a single failure of one offsite power source during non accident conditions. In the event the plant is in the 100/0 or 0/100 mode, a redundant source of offsite power can be obtained by backfeeding through the main transformer using a flexible connection that can be tied into the plant auxiliary transformer 11. The plant can survive on the available battery power, alternate power sources, and turbine driven Auxiliary Feedwater train during the estimated 8 hours required to provide this power transfer (Ref. 3). Therefore, the requirements of GDC 17 (Ref. 4) can be met at all times.

The AC, DC, and AC instrument bus electrical power distribution subsystems satisfy Criterion 3 of the NRC Policy Statement.

LCO

Train A and Train B of the AC, DC, and AC instrument bus electrical power distribution subsystems are required to be OPERABLE. The power distribution subsystems and their trains listed in Table B 3.8.9-1 ensure the availability of AC, DC, and AC instrument bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

(continued)

BASES (continued)

LCO
(continued)

OPERABLE AC, DC, and AC instrument bus electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages. Maintaining the Train A and Train B AC, DC, and AC instrument bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not compromised. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

Tie breakers between redundant safety related AC, DC, and AC instrument bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, which could cause the failure of a redundant subsystem and a loss of essential safety function(s).

If any of the following listed tie breakers are closed, the affected redundant electrical power distribution subsystem is considered inoperable. This does not, however, preclude AC buses from being powered from the same offsite circuit.

- a. AC power 480 V safeguards bus tie breakers (Ref. 5)

Bus-Tie 14-16
Bus-Tie 16-14
Bus-Tie 17-18
Bus-Tie 16-15
Bus-Tie 14-13

- b. DC control power automatic throwover switches (in normal position) (Ref. 6)

DG Control Panel A
DG Control Panel B
Bus 14 Control Power and Undervoltage Cabinet
Bus 16 Control Power and Undervoltage Cabinet
Bus 17 Control Power and Undervoltage Cabinet
Bus 18 Control Power and Undervoltage Cabinet

(continued)

BASES (continued)

(continued)

BASES (continued)

LCO ~~The trains as specified in Table 3.8.9-1 only identify the~~
~~(continued) major AC, DC, and AC instrument bus electrical power~~
~~distribution subsystem components.~~ ~~Technical Support Center battery~~
~~(continued) power Battery A and B (Ref. 6)~~
~~connections to DC~~

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TSC/Battery A Fused Disconnect Switch
TSC/Battery B Fused Disconnect Switch

~~The trains as specified in Table B 3.8.9-1 only identify the~~
~~major AC, DC, and AC instrument bus electrical power~~
~~distribution subsystem components.~~ A train is defined to
begin from the boundary of the power source for the
respective subsystem (as defined in the power source LCOs),
and continues up to the isolation device for the supplied
safety related or ESF component (e.g., safety injection
pump). The isolation device for the supplied safety related
or ESF component is only considered part of the train when
the device is not capable of opening to isolate the failed
component from the train (e.g., breaker unable to open an
overcurrent). Otherwise, the failure of the isolation
device to close to provide power to the component is
addressed by the respective component's LCO. The isolation
device for nonsafety related components are considered part
of the train since these devices must be available to
protect the safety related functions. Therefore, the train
boundary essentially ends at the motor control center or bus
which supplies multiple components.

The inoperability of any component within the above defined
train boundaries renders the train inoperable.

APPLICABILITY

The electrical power distribution subsystems are required to
be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant
pressure boundary limits are not exceeded as a result
of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment
OPERABILITY and other vital functions are maintained
in the event of a postulated DBA.

(continued)



BASES

APPLICABILITY (continued) Electrical power distribution subsystem requirements for MODES 5 and 6 are addressed in LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

ACTIONS

A.1

With one AC electrical power distribution train inoperable, the remaining AC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining AC power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels which comprise a train must be restored to OPERABLE status within 8 hours.

The worst case Condition A scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the plant is more vulnerable to a complete loss of AC power.

The Completion Time for restoring the inoperable train before requiring a plant shutdown is limited to 8 hours because of:

- a. The potential for decreased safety if the plant operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE train with AC power which results in the loss of multiple safety functions.

(continued)

BASES

ACTIONS
(continued)

B.1

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With one ~~required~~-AC instrument bus electrical power distribution train inoperable, the remaining OPERABLE AC instrument bus train is capable of supporting the minimum safety functions necessary to shut down the plant and maintain it in the safe shutdown condition. Overall reliability is reduced, however, because a single failure in the remaining AC instrument bus train could result in the minimum ESF functions not being supported. Therefore, the ~~required~~-AC instrument bus train must be restored to OPERABLE status within 2 hours.

Condition B represents one AC instrument bus train without power which includes the potential loss of both the DC source and the associated AC sources to the instrument bus. In this situation, the plant is significantly more vulnerable to a complete loss of all noninterruptible power. Therefore, the Completion Time is limited to 2 hours due to the potential vulnerabilities. Taking exception to LCO 3.0.2 for components without adequate 120 VAC power, that would have Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

- a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate 120 VAC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and
- c. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE AC instrument bus train.

(continued)

BASES

ACTIONS
(continued)

B.1 (continued)

The 2 hour Completion Time takes into account the importance to safety of restoring the AC instrument bus train to OPERABLE status, the redundant capability afforded by the other OPERABLE instrument bus train, and the low probability of a DBA occurring during this period.

C.1

With one DC electrical power distribution train inoperable, the remaining DC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required DC distribution panels must be restored to OPERABLE status within 2 hours.

Condition C represents one train without adequate DC power (e.g., the battery and required battery charger are inoperable). In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. Therefore, the Completion Time is limited to 2 hours due to this potential vulnerability. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in plant conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

(continued)

BASES

ACTIONS

C.1 (continued)

- c. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE train with DC power.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

(212) ~~If with two or more trains are with inoperable electrical power distribution subsystems,~~ the potential for a loss of safety function is greater. If a loss of safety function exists, no additional time is justified for continued operation and LCO 3.0.3 must be entered. This Condition may be entered with the loss of two trains of the same electrical power distribution subsystem, or with loss of Train A of one electrical power distribution subsystem coincident with the loss of Train B of a second electrical power distribution subsystem such that a loss of safety function exists.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.9.1

(212)

This SR verifies that the AC, DC, and AC instrument bus electrical power distribution subsystems ~~trains~~ are functioning properly, with all required power source circuit breakers closed, tie-breakers open, and the buses energized from their allowable power sources. Required voltage for the AC electrical power distribution subsystem is ≥ 420 VAC; for the DC electrical power distribution subsystem, ≥ 108.6 VDC; and for AC instrument bus electrical power distribution subsystem, between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the redundant capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. UFSAR, Section 8.3.1.
4. 10 CFR 50, Appendix A, GDC 17.
5. UFSAR, Figure 8.3-1.
6. UFSAR, Figure 8.3-6.
7. ~~UFSAR, Figure 8.3-4.~~

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Table B 3.8.9-1 (page 1 of 1)
 AC and DC Electrical Power Distribution Systems

DISTRIBUTION SUBSYSTEM	VOLTAGE	TRAIN A	TRAIN B
AC Power	480 V	Bus 14 Bus 18	Bus 16 Bus 17
DC Power	125 V	Main DC Fuse Cabinet A (DCPDPCB02A) Main DC Distribution Panel A (DCPDPCB03A) Aux Bldg DC Distribution Panel A (DCPDPA01A) Aux Bldg DC Distribution Panel A1 (DCPDPA02A) DG A DC Distribution Panel A (DCPDPOG01A) Screenhouse DC Distribution Panel A (DCPDPSH01A) MCB DC Distribution Panel A (DCPDPCB04A)	Main DC Fuse Cabinet B (DCPDPCB02B) Main DC Distribution Panel B (DCPDPCB03B) Aux Bldg DC Distribution Panel B (DCPDPA01B) Aux Bldg DC Distribution Panel B1 (DCPDPA02B) DG B DC Distribution Panel B (DCPDPOG01B) Screenhouse DC Distribution Panel B (DCPDPSH01B) MCB DC Distribution Panel B (DCPDPCB04B) Turbine Bldg DC Distribution Panel (DCPDPTB01B)
AC Instrument Bus	120 V	Bus A Bus B	Bus C

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems—MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

(109) In MODES 5 or 6, the number of required AC, DC, and AC instrument bus electrical power distribution subsystems, or the number of required trains within these electrical power distribution subsystems may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the electrical power distribution subsystems, must be removed from service.

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum AC, DC, and AC instrument bus electrical power distribution subsystems during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition and refueling condition.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. Therefore, the OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution subsystems ensures that one train of the onsite power or offsite AC sources are OPERABLE in the event of:

- a. An assumed loss of all offsite AC power;
- b. An assumed loss of all onsite standby AC power; or
- c. A worst case single failure.

This reduction in required AC sources is allowed because many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary (reactor coolant temperature and pressure) and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and result in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for the systems required in MODES 5 and 6.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6 this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC, DC, and AC instrument bus electrical power distribution subsystems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of AC, DC, and AC instrument bus electrical power distribution subsystems, trains within these subsystems, and equipment and components within these trains are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

The LCOs which apply when the Reactor Coolant System is $\leq 200^\circ\text{F}$ and which may require a source of electrical power are:

- LCO 3.1.1 SHUTDOWN MARGIN (SDM)
- LCO 3.3.1 Reactor Trip System (RTS) Instrumentation
- (167) LCO ~~3.3.53~~ 3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation
- LCO ~~3.3.73~~ 3.5/6 Control Room Emergency Air Treatment System (CREATS) Actuation
- LCO 3.4.7 RCS Loops - MODE 5, Loops Filled
- LCO 3.4.8 RCS Loops - MODE 5, Loops Not Filled
- LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System
- LCO 3.7.9 Control Room Emergency Air Treatment System (CREATS)
- (167) LCO 3.9.2 Nuclear Instrumentation
- LCO ~~3.9.3~~ Residual Heat Removal (RHR) and Coolant Circulation - High Water Level
- ~~LCO-3.9.4~~ Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft
- ~~LCO 3.9.5~~ Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level Water Level < 23 Ft

(212) Maintaining these portions of the necessary trains of the AC, DC, and AC instrument bus electrical power the distribution subsystems energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

(continued)

BASES (continued)

APPLICABILITY The AC, DC, and AC instrument bus electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a postulated event and maintain the plant in the cold shutdown or refueling condition are available.

The AC, DC, and AC instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9, "Distribution Systems—MODES 1, 2, 3, and 4."

ACTIONS

A.1

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations involving positive reactivity additions. By allowing the option to declare required features associated with an inoperable distribution subsystem or train inoperable, appropriate restrictions are implemented in accordance with the LCO ACTIONS of the affected required features.

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

With one or more required electrical power distribution subsystems or trains inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position of normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

(212)

(continued)

BASES (continued)

ACTIONS A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

It is further required to immediately initiate action to restore the required AC, DC, and AC instrument bus electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

In addition to performance of the above conservative Required Actions, a required residual heat removal (RHR) loop may be inoperable. In this case, Required Actions A.2.1, A.2.2, A.2.3, and A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 requires declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.10.1

(212) This Surveillance verifies that the AC, DC, and AC instrument bus electrical power distribution subsystems are functioning properly, with all the required power source circuit breakers closed, required tie-breakers open, and the required buses energized from their allowable power sources. Required voltage for the AC power distribution electrical subsystem is ≥ 420 VAC, for the DC power distribution electrical subsystem ≥ 108.6 VDC, and for AC instrument bus power distribution electrical subsystem is between 113 VAC and 123 VAC. Required voltage for the twinco panels supplied by the 120 VAC instrument buses is between 115.6 VAC and 120.4 VAC. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The Frequency of 7 days takes into account the capability of the AC, DC, and AC instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

None.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

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APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in the COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux channels ~~monitors~~ shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One source range neutron flux channel monitor inoperable.</p>	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
<p>B. Two source range neutron flux channels monitors inoperable.</p>	B.1 Initiate action to restore one source range neutron flux channel monitor to OPERABLE status.	Immediately
	AND	
	B.2 Suspend CORE ALTERATIONS Perform SR 3.9.1.1.	Immediately 4 hours
	AND	
	B.3 Suspend positive reactivity additions.	Immediately
	AND	
	B.4 Perform SR 3.9.1.1.	Once per 12 hours thereafter

11/11/11



11/11/11

CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)		
C. No audible count rate.	C.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	C.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	C.3 Perform SR 3.9.1.1	4 hours <u>AND</u> Once per 12 hours thereafter

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.2.1 Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2 -----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION of each source range.	24 months

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3.9 REFUELING OPERATIONS

3.9.3 ~~Residual Heat Removal (RHR) and Coolant Circulation Water Level \geq 23 Ft~~
~~Containment Penetrations~~

LCO 3.9.3 ~~One RHR loop~~ ~~The containment penetrations shall be OPERABLE and in operation~~ ~~the following status:~~

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- a. ~~The equipment hatch shall be either:~~
 - 1. ~~held in place with at least one access door closed,~~ ^{bolted} ~~or~~
 - 2. ~~isolated by a closure plate that restricts air flow from containment;~~
- b. ~~One door in the personnel air lock~~ ^{shall be} ~~closed; and~~
- c. ~~Each penetration providing direct access from the containment atmosphere to the outside atmosphere~~ ^{shall be} ~~either:~~
 - 1. ~~closed by a manual or automatic isolation valve, blind flange, or equivalent, or~~
 - 2. ~~capable of being closed by an OPERABLE Containment Ventilation Isolation System.~~

~~APPLICABILITY: During CORE ALTERATIONS,~~
~~During movement of irradiated fuel assemblies within~~
~~containment.~~

~~ACTIONS~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A: One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately



SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2 Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

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3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft

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ECO 3.9.4 One RHR loop shall be OPERABLE and in operation.

-----NOTE-----
The required RHR loop may be removed from operation for \leq 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System (RCS) boron concentration.

APPLICABILITY: MODE 6 with the water level \geq 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend all operations involving a reduction in RCS boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	

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CONDITION	REQUIRED ACTION	COMPLETION TIME
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~~SURVEILLANCE (continued)~~

~~SURVEILLANCE REQUIREMENTS~~

~~FREQUENCY A.4 (continued)~~

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~~12 hours
SR 3.9.3.1
Verify one OPERABLE RHR loop is in operation.
A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.~~

~~3.9 REFUELING OPERATIONS
3.9.4 Residual Heat Removal (RHR) and Coolant Circulation Water Level $<$ 23 Ft
LCO 3.9.4
Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation. 4 hours~~

~~SURVEILLANCE REQUIREMENTS~~

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SURVEILLANCE	FREQUENCY
SR 3.9.4.1 Verify one RHR loop is in operation and circulating reactor coolant.	12 hours

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

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LCO 3.9.5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One less than the required number of RHR loop inoperable loops OPERABLE.	A.1 Initiate action to restore RHR loop loop(s) to OPERABLE status.	Immediately
	OR A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

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~~B. Two RHR loops inoperable.~~
 OR
~~No RHR loop in operation.~~

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~~B.1 Suspend all operations involving reduction in Reactor Coolant System boron concentration.~~

Immediately

AND

~~B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.~~

Immed ~~(abandoned)~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>SURVEILLANCE REQUIREMENTS B: No RHR loop in operation.</p>	<p>B.1 Suspend operations involving a reduction in Reactor Coolant System boron concentration.</p> <p>AND</p> <p>B.2 Initiate action to restore one RHR loop to operation.</p> <p>AND</p> <p>B.3 Close all containment penetrations providing direct access from containment to outside atmosphere.</p>	<p>SURVEILLANCE Imme diately</p> <p>Immediately</p> <p>4 hours</p>

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~~FREQUENCY SURVEILLANCE REQUIREMENTS~~

~~SR 3.9.4.1 - Verify one OPERABLE RHR loop is in operation.~~

<p>12 hours SURVEILLANCE</p>	<p>SR-3.9.4.2 Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation. FREQUENCY</p>
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SR 3.9.5.1 Verify one RHR loop is in operation and circulating reactor coolant. 12 hours

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SR 3.9.5.2 Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation. 7 days

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3.9 REFUELING OPERATIONS

3.9.5.13.9.6 Refueling Cavity Water Level

(221)

LCO 3.9.5.13.9.6 Refueling cavity water level shall be maintained \geq 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment,
During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.5.13.9.6.1 Verify refueling cavity water level is \geq 23 ft above the top of reactor vessel flange.	24 hours

(221)

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration ensures the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the filled portions of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity that are hydraulically coupled to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant. The refueling boron concentration limit is specified in the Core Operation Limits Report (COLR). Plant refueling procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant refueling procedures.

Atomic Industrial Forum (AIF) GDC 27 requires that two independent reactivity control systems preferably of different design principles be provided (Ref. 1). In addition to the reactivity control achieved by the control rods, reactivity control is provided by the chemical and volume control system (CVCS) which regulates the concentration of boric acid solution neutron absorber (neutron absorber) in the RCS. The CVCS is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes which may stress or damage the fuel beyond allowable limits.

The reactor is brought to shutdown conditions (i.e., MODE 5) before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized the vessel head is unbolted and removed. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by use of the Residual Heat Removal (RHR) System pumps.

(continued)

BASES

BACKGROUND

The pumping action of the ~~Residual Heat Removal (RHR)~~RHR System

(continued)
(continued)

into the RCS, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity provide mixing for the borated coolant in the refueling canal.

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The RHR System is in operation during refueling (see LCO 3.9.3, "~~Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft.~~" and LCO 3.9.4, "~~Residual Heat Removal (RHR) and Coolant Circulation - Water Level \geq 23 Ft.~~" and LCO 3.9.5, "~~Residual Heat Removal (RHR) and Coolant Circulation - Water Level $<$ 23 Ft.~~") to provide forced circulation in the RCS and assist in maintaining the boron concentration in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

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APPLICABLE
SAFETY ANALYSES

During refueling operations, two types of accidents can occur within containment that affect the fuel and require control of reactivity. These two accident types are a fuel handling accident and a boron dilution event. Both accidents assume that initial core reactivity is at its highest (i.e., at the beginning of the fuel cycle or the end of refueling).

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A fuel handling accident ~~occurs~~can occur during fuel movement in the reactor vessel, the refueling canal, or the refueling cavity and includes a dropped fuel assembly and an incorrectly transferred fuel assembly. The most limiting fuel handling accident is a dropped fuel assembly which is dropped adjacent to other fuel assemblies such that it results in the largest exposure of fuel in the dropped assembly. The negative reactivity effect of the soluble boron compensates for the increased reactivity for both types of accidents. Hence, the boron concentration ensures that $k_{\text{eff}} \leq 0.95$ (i.e., 5% $\Delta k/k$ SHUTDOWN MARGIN) during the refueling operation.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The second type of accident is a boron dilution event which results from inadvertent addition of unborated water to the RCS, refueling cavity, and refueling canal. The assumptions used in the boron dilution event (Ref. 2) provide for a maximum dilution flow of 120 gpm through two charging pumps (i.e., 60 gpm per pump) using unborated water as supplied by the two reactor makeup water pumps (60 gpm per pump). The RCS is also assumed to be at low water levels, uniformly mixed by the RHR System, with the minimum boron concentration as specified in the COLR. The operator has prompt and definite indication of significant boron dilution from an audible count rate function provided by the source range neutron flux instrumentation (see LCO 3.9.2, "Nuclear Instrumentation"). The increased count rate is a function of the effective subcritical multiplication factor. The results of this analysis conclude that an operator has at least 48.8 minutes before SHUTDOWN MARGIN is lost and the reactor goes critical which is sufficient time for operators to mitigate this event. This time is also greater than the 30 minutes required by Reference 3 for dilution events during refueling. Isolating the boron dilution source is performed by closing valves and/or stopping the reactor makeup water pumps.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

(194)

The LCO requires that a minimum boron concentration be maintained in the ~~portions of the RCS, the refueling canal, and the refueling cavity~~ and the portions of the RCS that are hydraulically coupled with the reactor core while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{off} of ≤ 0.95 is maintained during fuel handling operations and that a core k_{off} of < 1.0 is maintained during a boron dilution event. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

(continued)

BASES (continued)

APPLICABILITY

(194)

(195)

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{\text{off}} \leq 0.95$ during fuel handling operations. In MODES 1 and 2 with $k_{\text{off}} \geq 1.0$, LCO 3.1.4, "Rod Group Alignment Limits Limit," LCO 3.1.5, "Shutdown Bank Insertion Limits Limit," and LCO 3.1.6, "Control Bank Insertion Limits" ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 2 with $k_{\text{off}} < 1.0$ and MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures an adequate amount of negative reactivity is available to maintain the reactor subcritical.

ACTIONS

A.1, A.2, and A.3

If the boron concentration of the filled portions of the RCS, the refueling canal, and the refueling cavity hydraulically coupled to the reactor core, is less than its limit, an inadvertent criticality may occur due to a boron dilution event or incorrect fuel loading. To minimize the potential of an inadvertent criticality resulting from a fuel loading error or an operation that could cause a reduction in boron concentration, CORE ALTERATIONS and positive reactivity additions must be suspended immediately.

(194)

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control (i.e., other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures) shall not preclude moving a component to a safe position.

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

There are no safety analysis assumptions of boration flow rate and concentration that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

(continued)

BASES (continued)

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Once action has been initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE
REQUIREMENTS

SR 3.9.1.1

(194) This SR ensures the coolant boron concentration of the refueling canal, the refueling cavity, and the portions of the RCS that are hydraulically coupled, is within the COLR limits. The boron concentration of the coolant is determined by chemical analysis. The sample should be representative of the portions of the RCS, the refueling canal, and the refueling cavity that are hydraulically coupled with the reactor core.

A Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 27, Issued for comment July 10, 1967.
 2. UFSAR, Section 15.4.4.2.
 3. NUREG-0800, Section 15.4.6.
-

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux channels~~monitors~~ are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux channels~~monitors~~ (N-31 and N-32) are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

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The installed source range neutron flux detectors are proportional counters that are filled with boron trifluoride (BF₃) gas (Ref. 1). The detectors monitor the neutron flux in counts per second and provide continuous visual indication in the control room. Audible count rate is also available in the control room from either of the source range neutron flux channels~~monitors~~ to alert operators to a possible boron dilution event. The NIS is designed in accordance with the criteria presented in Reference 2.

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APPLICABLE

Two OPERABLE source range neutron flux channels~~monitors~~ are required

SAFETY ANALYSES

to provide redundant indication to alert operators of unexpected changes in core reactivity. An increase in the audible count rate alerts the operators that a boron dilution event is in progress. Sufficient time is available for the operator to recognize the increase in audible count rate and to terminate the event prior to a loss of SHUTDOWN MARGIN (see Bases for LCO 3.9.1, "Boron Concentration"). Isolating the boron dilution source is performed by closing valves and stopping reactor makeup water pumps.

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The source range neutron flux channels~~monitors~~ satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

(195)

This LCO requires two source range neutron flux channels ~~monitors~~ be OPERABLE to ensure redundant monitoring capability is available to detect changes in core reactivity.

To be OPERABLE, each ~~channel~~ ^{monitor} must provide visual indication and at least one of the two ~~channels~~ ^{monitors} must provide an audible count rate function in the control room.

(194)

~~With the discharge of fuel from core positions adjacent to source range detector locations, counts decreasing to zero is the expected response. Based on this indication alone, source range detection should not be considered inoperable. Following a full core discharge, source range response is verified with the initial fuel assembly reload.~~

APPLICABILITY

(195)

In MODE 6, the source range neutron flux ~~channels~~ ^{monitor} must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity conditions in this MODE. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

(195)

With only one source range neutron flux channel ~~monitor~~ OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions A.1 and A.2 shall not preclude completion of movement of a component to a safe position ~~or normal cooldown of the coolant volume for the purpose of system temperature control~~.

(194)

(194)

~~For the purpose of this Condition, a source range neutron flux channel is inoperable when no visual indication is available. The loss of the audible count rate function is addressed by Condition C, other than normal cooldown of the coolant volume for the purpose of system temperature control within established procedures).~~

(continued)

BASES

B.1 and B.2

195

With no source range neutron flux monitor OPERABLE there are no direct means of detecting changes in core reactivity. Therefore, actions to restore a monitor to OPERABLE status shall be initiated immediately and continue until a source range neutron flux monitor is restored to OPERABLE status.

(continued)

BASES

ACTIONS

B.1, and B.2 (continued)

194
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~~With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made per Required Actions A.4~~

(continued)

~~With no source range neutron flux channel OPERABLE there are no direct means of detecting changes in core reactivity] and A. Therefore, actions to restore a channel to OPERABLE status shall be initiated immediately and continue until a source range neutron flux channel the core reactivity condition is restored to OPERABLE status stabilized until the source range neutron flux monitors are OPERABLE.~~

~~Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux channels are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure the required boron concentration exists.~~

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~~The Completion Time of once per 12 hours is sufficient to obtain and analyze coolant samples for boron concentration and to ensure unplanned changes in boron concentration would be identified. The 12-hour Completion Time is reasonable, considering the low probability frequency of a change once per 12 hours ensures unplanned changes in core reactivity during this time period boron concentration would be identified. The 12 hour frequency is reasonable, considering the low probability of a change in core reactivity during this time period.~~

C.1, C.2, and C.3

194
~~With no audible count rate available, only visual indication is available and prompt and definite indication of a boron dilution event has been lost. Therefore, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Actions C.1 and C.2 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.~~

that is a

(continued)



Since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the audible count rate capability is restored. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

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The Completion time of ~~once per 124~~ hours is sufficient to obtain and analyze coolant samples for boron concentration and ~~to ensure unplanned changes in boron concentration would be identified.~~

(continued)

~~ACTIONS~~ — The 12 hour Completion Time is reasonable, considering the
~~(continued)~~ — low probability frequency of a change once per 12 hours
ensures unplanned changes in core reactivity during this
time period boron concentration would be identified.

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~~SURVEILLANCE~~ — SR 3.9.2.1
~~REQUIREMENTS~~

~~SR 3.9.2.1 is the performance of a CHANNEL CHECK. The 12 hour
Frequency is reasonable, which is a comparison considering
the low probability of the parameter indicated on one
channel to a similar parameter on another channel a change in
core reactivity during this time period.~~

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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS SR 3.9.2.1

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This SR is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one monitor to a similar parameter on another monitor. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels monitors, but each channel monitor should be consistent with its local conditions.

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~~The inoperability of one source range neutron flux channel prevents performance of a CHANNEL CHECK for the operable channel. However, the Required Actions for the inoperable channel requires suspension of CORE ALTERATIONS and positive reactivity addition such that the CHANNEL CHECK of the operable channel can consist of ensuring consistency with known core conditions.~~

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

SR 3.9.2.2

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~~SR 3.9.2.2~~ This SR is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux channels monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. REFERENCES 1.—UFSAR, Section 7.7.3.2.
2. Atomic Industrial Forum (AIF) GDC 13 and 19, Issued for Comment July 10, 1967.

221 → Entire Section

B 3.9.3

B-3.9 REFUELING OPERATIONS

B 3.9.3 ~~Residual Heat Removal (RHR) and Coolant Circulation Water Level~~
~~> 23 Ft Containment Penetrations~~

In MODE 5, there are no accidents of concern which require containment.

BASES

BACKGROUND

~~The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and to provide mixing of the boric acid coolant to prevent thermal and boron stratification. (Re) release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.~~

~~The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.~~

~~The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place. Good engineering practice dictates that the bolts required to hold the equipment hatch in place be~~

bolted →

and that the bolts

a minimum of 4 bolts be used

(continued)

10/10/10



10/10/10

BASES

221

approximately equally spaced. As an alternative, the equipment hatch can be isolated by a closure plate that restricts air flow from containment.

(continued)

BASES

BACKGROUND
(continued)

221

The containment equipment and personnel air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of plant shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed in the personnel and equipment hatch (unless the equipment hatch is isolated by a closure plate).

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The Shutdown Purge System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a Mini-Purge System, includes a 6 inch purge penetration and a 6 inch exhaust penetration. During MODES 1, 2, 3, and 4, the shutdown purge and exhaust penetrations are isolated by a blind flange with two O-rings that provide the necessary boundary. The two air operated valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation Instrumentation System. Neither of the subsystems is subject to a Specification in MODE 5.

(continued)

BASES

BACKGROUND
(continued)

221

In MODE 6, large air exchangers are used to support refueling operations. The normal 36 inch Shutdown Purge System is used for this purpose, and each air operated valve is closed by the Containment Ventilation Isolation Instrumentation in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation."

The Mini-Purge System also remains operational in MODE 6, and all four valves are also closed by the Containment Ventilation Isolation Instrumentation.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE
SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed using the criteria of Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in 10 CFR 100. Standard Review Plan (SRP), Section 15.7.4, Rev. 1 (Ref. 2), requires containment closure even though this is not an assumption of the accident analyses. The acceptance limits for offsite radiation exposure is 96 rem (Ref. 3).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement since these are assumed in the SRP.

(continued)

221

LCO This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that at least one valve in each of these penetrations is isolable by the Containment Ventilation Isolation System.

APPLICABILITY The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions, no requirements are placed on containment penetration status.

ACTIONS A.1 and A.2

If the containment equipment hatch (or its closure plate), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

SURVEILLANCE REQUIREMENTS SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

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The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. ~~A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.~~ As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

(continued)

REFERENCES

1. UFSAR, Section 15.7.
2. NUREG-800, Section 15.7.4, Rev. 1, July 1981.
3. Letter from D. M. Crutchfield, NRC, to J. Maier, RG&E,
Subject: "Fuel Handling Accident Inside Containment,"
dated October 7, 1981.
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B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation—Water Level
 \geq 23 Ft

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BASES

BACKGROUND: The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), and to provide mixing of the borated coolant to prevent thermal and boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s) where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS loop "B" cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSES

The safety analysis for the boron dilution event during refueling assumes one RHR loop is in operation (Ref. 2). This initial assumption ensures continuous mixing of the borated coolant in the reactor vessel. The analysis also assumes the RCS is at equilibrium boron concentration and dilution occurs uniformly throughout the system. Therefore, thermal or boron stratification is not postulated. In order to ensure adequate mixing of the borated coolant, one loop of the RHR System is required to be OPERABLE, and in operation while in MODE 6, with water level \geq 23 ft above the top of the reactor vessel flange.

(continued)



BASES

APPLICABLE
SAFETY ANALYSES
(continued)

(194)

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. Due to the water volume available in the RCS with a water level \geq 23 ft above the top of the reactor vessel flange, a significant amount of time exists before boiling of the coolant would occur following a loss of the required RHR pump. Since the loss of the required RHR pump results in the ~~suspension of requirement to suspend operations involving a reduction in reactor coolant boron concentration, a boron dilution event is very unlikely.~~ Therefore, this requirement dictates that single failures are not considered for this LCO due to the time available to operators to respond to a loss of the operating RHR pump.

(194)

The LCO permits de-energizing the required RHR pump for short durations, ~~under the condition provided no operations are permitted that they would cause a reduction in the RCS boron concentration is not reduced.~~ This conditional de-energizing of the required RHR pump does not result in a challenge to the fission product barrier or result in coolant stratification.

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(194)

~~Residual Heat Removal (RHR) RHR and Coolant Circulation—High Water Circulation—Water Level \geq 23 Ft~~ satisfies criterion 42 of the NRC Policy—Statement.

LCO

(194)

Only one RHR loop is required for decay heat removal in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. One RHR loop is required to be OPERABLE and in operation to provide mixing of borated coolant ~~to minimize the possibility of criticality.~~

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg.

(continued)



BASES

LCO
(continued)

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This allows the operator to view the core and permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles. This also permits operations such as RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity. Should both RHR loops become inoperable at anytime during operation in accordance with this Note, the Required Actions of this LCO should be immediately taken.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.5, "Refueling Cavity Water Level."

Requirements for the RHR System in MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.4, "RCS Loops—MODE 1 $>$ 8.5% RTP;" LCO 3.4.5, "RCS Loops—MODES 1 \leq 8.5% RTP, 2 and 3;" LCO 3.4.6, "RCS Loops—MODE 4;" ~~LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled;"~~ ~~LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled;"~~ and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." The RHR loop requirements in MODE 6 with the water level $<$ 23 ft are located in LCO ~~3.9.4~~ 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Water Level $<$ 23 Ft."

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(continued)

BASES (continued)

ACTIONS

A.1, A.2, and A.3

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with a lower boron concentration than that contained in the RCS. Therefore, actions that could result in a reduction in the coolant boron concentration must be suspended immediately.

With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition. Therefore, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core.

(continued)

BASES (continued)

With the plant in MODE 6 and the refueling water level \geq 23 ft above the top of the reactor vessel flange, removal of decay heat is by ambient losses only. Therefore, corrective actions shall be initiated immediately and shall continue until RHR loop requirements are satisfied.

SURVEILLANCE
REQUIREMENTS

SR 3.9.3.1

~~This SR requires verification every 12 hours that one OPERABLE RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is~~

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~~If RHR loop requirements are not met, all containment penetrations providing decay heat removal capability and mixing of the borated coolant direct access from the containment atmosphere to prevent thermal and boron stratification in the core the outside atmosphere must be closed within 4 hours. The frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor the RHR loop performance. If requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere.~~

REFERENCES

~~Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.~~

221
~~UFSAR The Completion Time of 4 hours is reasonable, Section 5.4.5 based on the low probability of the coolant boiling in that time.~~

(continued)

BASES (continued)

~~2~~SURVEILLANCE SR 3.9.4.1
REQUIREMENTS

This SR requires verification every 12 hours that one RHR loop is in operation and circulating reactor coolant.

UFSAR, Section 15.4.4.2 Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core.

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BASES (continued)

~~RHR and Coolant Circulation Water Level $<$ 23 Ft
B 3.9.4~~

~~B 3.9 REFUELING OPERATIONS~~

~~B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation Water Level
 $<$ 23 Ft~~

BASES

BACKGROUND — The purpose of the RHR System in MODE 6/2 hours is to remove decay heat sufficient considering other indications and sensible heat from alarms available to the Reactor Coolant System (RCS), and to provide mixing of the borated coolant to prevent thermal and boron stratification (Refoperator in the control room to monitor RHR loop performance).

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- REFERENCES
1. UFSAR, Section 5.4.5.
 2. UFSAR, Section 15.4.4.2.
-

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

BASES

BACKGROUND The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), and to provide mixing of the borated coolant to prevent thermal and boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s) where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS loop "B" cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and bypass line(s). Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

The safety analysis for the boron dilution event during refueling assumes one RHR loop is in operation (Ref. 2). This initial assumption ensures continuous mixing of the borated coolant in the reactor vessel. The analysis also assumes the RCS is at equilibrium boron concentration and dilution occurs uniformly throughout the system. Therefore, thermal or boron stratification is not postulated.

While there is no explicit analysis assumption for the decay heat removal function of the RHR System in MODE 6, if the reactor coolant temperature is not maintained, boiling of the coolant could result. This could lead to a loss of coolant in the reactor vessel. In addition, boiling of the coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of coolant and the reduction of boron concentration in the reactor coolant could eventually challenge the integrity of the fuel cladding, which is a fission product barrier.

(continued)

BASES



(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In order to prevent a challenge to fuel cladding and to ensure adequate mixing of the borated coolant, two loops of the RHR System are required to be OPERABLE, and one loop in operation while in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange.

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~~Residual Heat Removal (RHR) RHR and Coolant Circulation-Low Water Circulation-Water Level < 23 Ft~~ satisfies criterion 4 of the NRC Policy Statement.

LCO

Both RHR loops must be OPERABLE in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange. In addition, one RHR loop must be in operation in order to remove decay heat and provide mixing of borated coolant to minimize the possibility of criticality.

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An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path. The flow path starts in the RCS loop "A" hot leg and is returned to the RCS loop "B" cold leg.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant.

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Requirements for the RHR System in MODES 1, 2, 3, 4, and 5 are covered by LCO 3.4.4, "RCS Loops-MODE 1 > 8.5% RTP;" LCO 3.4.5, "RCS Loops-MODES 1 ≤ 8.5% RTP, 2 and 3;" LCO 3.4.6, "RCS Loops-MODE 4;" ~~LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled;"~~ LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled;" and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." The RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO ~~3.9.3~~ 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-Water Level ≥ 23 Ft."

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

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If only one RHR loop is OPERABLE and in operation, less than the required number of RHR loops are OPERABLE, redundancy for RHR is lost. Action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and operation or until ≥ 23 ft of water level is established above the reactor vessel flange. Action must be initiated to restore either a second loop to OPERABLE status or when the water level ≥ 23 ft above the top of reactor vessel flange, the Applicability changes to that of LCO 3.9.4, and only one RHR loop is required to be OPERABLE and in operation. The An immediate Completion Time reflects the importance of maintaining the availability of two paths necessary for heat removal. An operator to initiate corrective actions.—

The actions to restore must continue until either the second RHR loop is restored to OPERABLE status or water level is established ≥ 23 ft above the top of reactor vessel flange.

1 and B.1 and B.2

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If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. 2

If no RHR loop is in operation or if no loop is OPERABLE all operations involving a reduction in boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. The potential for reduced boron concentrations by the addition of RCS water with a lower boron concentration must be reduced than that contained in the RCS must be reduced to prevent a criticality event. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality. Therefore, forced circulation is required to provide proper mixing and preserve the margin to criticality. Operations involving a reduction in this type of operation RCS boron concentration must be suspended immediately. The immediate Completion Time reflects the importance of maintaining RCS boron concentration. Actions shall also be initiated immediately, and continued, to restore one RHR loop to operation for heat removal. The action to restore must continue until one loop is restored to in Conditions A and B concurrently, the restoration of two

(continued)

BASES (continued)

~~OPERABLE status and operation RHR loops and one operating RHR loop should be accomplished expeditiously.~~

SURVEILLANCE
REQUIREMENTS

SR 3.9.4.1

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~~This SR requires verification every 12 hours that one OPERABLE RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow~~

~~If no RHR loop is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. The frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor. With the RHR loop performance requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere.~~

(continued)

BASES (continued)

~~SURVEILLANCE~~ — ~~SR 3.9.4.2~~
~~REQUIREMENTS~~

~~(continued)~~

~~Verification of closing containment penetrations that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, are open to maintain decay heat removal and reactor coolant circulation the outside atmosphere ensures that dose limits are not exceeded.~~

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~~The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS SR 3.9.5.1

(195) This SR requires verification every 12 hours that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. The frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.9.5.2

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the standby pump. The frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. UFSAR, Section 5.4.5.
 2. UFSAR, Section 15.4.4.2.
-

B 3.9 REFUELING OPERATIONS

221 B 3.9.53.9.6 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, requires a minimum water level of 23 ft above the top of the reactor vessel flange. This requirement ensures a sufficient level of water is maintained in the refueling cavity or portions hydraulically connected (e.g., refueling canal) to retain iodine fission product activity resulting from a fuel handling accident in containment (Ref. 1). The retention of iodine activity by the water limits the offsite dose from the accident well within the values specified in 10 CFR 100 (Ref. 2).

APPLICABLE
SAFETY ANALYSES

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 100 to be used in the accident analysis for iodine (Ref. 3). This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 3).

With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 2).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits and preserves the assumptions of the fuel handling accident analysis (Ref. 1). As such, it is the minimum required level during movement of fuel assemblies within containment. Maintaining this minimum water level in the refueling cavity also ensures that ≥ 23 ft of water is available in the spent fuel pool during fuel movement assuming that containment and Auxiliary Building atmospheric pressures are equal.

APPLICABILITY This LCO is applicable when moving irradiated fuel assemblies within containment. This LCO is also applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts. The LCO ensures a sufficient level of water is present in the refueling cavity to minimize the radiological consequences of a fuel handling accident in containment. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.11, "Spent Fuel Pool (SFP) Water Level."

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ACTIONS A.1 and A.2

When the initial condition assumed in the fuel handling accident cannot be met, steps should be taken to preclude the accident from occurring. With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

221
SURVEILLANCE
REQUIREMENTS

SR ~~3.9.5.13.9.6.1~~

Verification of a minimum refueling cavity water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 15.7.3.3.
 2. 10 CFR 100.
 3. Regulatory Guide 1.25.
-

4.0 DESIGN FEATURES

4.1 Site Location

The site for the R.E. Ginna Nuclear Power Plant is located on the south shore of Lake Ontario, approximately 16 miles east of Rochester, New York.

4.2 ~~Reactor Core~~

4.2.1 ~~Fuel Assemblies~~

The ~~reactor~~ exclusion area boundary distances from the plant shall contain ~~121 fuel assemblies~~ be as follows:

<u>Direction</u>	<u>Distance (m)</u>
N (including offshore)	8000
NNE	8000
NE	8000
ENE	8000
E	747
ESE	640
SE	503
SSE	450
S	450
SSW	450
SW	503
WSW	915
W	945
WNW	701
NW	8000
NNW	8000

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4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of ~~zirconium alloy~~ zirconium alloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO₂) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in

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(continued)

4.0 DESIGN FEATURES

accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

(continued)

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be as approved by the NRCs ~~silver~~ indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;
- b. $k_{\text{eff}} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c.

Consolidated rod storage canisters containing ~~≤ 358 undamaged rods or ≤ 110 bowed, broken, or otherwise failed rods~~ may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. ~~Canisters must meet the~~ However, the consolidated rod storage canister located in Region RGAF2 may exceed these requirements of Specification 3.7.17 (for the applicable region only) and Specifications 4.3.1.1. a and 4.3.1.1. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be ≤ 2150 BTU/hr. b-

4.3.1.2 The new fuel storage dry racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

(continued)



4.0 DESIGN FEATURES (continued)

- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

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- c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.2 Drainage

226 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool to ~~< 23 ft above the top of the fuel assemblies pool~~ below elevation 257'0" (mean sea level).

4.3 ~~Fuel Storage (continued)~~

4.3.3 Capacity

226 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 2032 ~~1016~~ fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ~~Plant Manager~~ ~~plant manager~~ shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

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The ~~Plant Manager~~ ~~plant manager~~, or his designee, shall approve prior to implementation, each proposed test, experiment or modification to structures, systems or components that affect nuclear safety.

5.1.2 The Shift Supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the plant is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the plant is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the UFSAR;

b. ~~The Plant Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; and~~

~~ereport to the corporate vice president specified in 5.2.1.~~

~~A specified senior corporate executive shall have corporate responsibility, shall be responsible for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.~~

c.

A corporate vice president shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

(continued)



5.2 Organization (continued)

- (153) d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

5.2 Organization (continued)

5.2.2 Plant Staff

The plant staff organization shall include the following:

- a. An auxiliary operator shall be assigned to the shift crew with fuel in the reactor. An additional auxiliary operator shall be assigned to the shift crew while the plant is in MODE 1, 2, 3 or 4.

~~5.2.2 Plant Staff (continued)~~

- b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.e for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.
- c. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
- d. The amount of overtime worked by plant staff members performing safety related functions shall be limited and controlled in accordance with a NRC approved program.

- e. ~~The operations manager or operations middle manager shall hold a SRO license.~~

- f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant. The STA shall be assigned to the shift crew while the plant is in MODE 1, 2, 3 or 4 and shall meet the qualifications ~~specified within a NRC approved~~ STA training program.

Contained in the

Specified in VFFAR
13.2

Specified in plant procedures. Changes to the guidelines in these procedures shall be submitted to the NRC for review.

5.0 ADMINISTRATIVE CONTROLS

5.3 Plant Staff Qualifications

- 5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI Standard N18.1-1971, as supplemented by Regulatory Guide 1.8, Revision 1, September 1975, for comparable positions.
-

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
 - c. Effluent and environmental monitoring;
 - d. Fire Protection Program implementation; and
 - e. All programs specified in Specification 5.5.
-

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),
 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the onsite review function and the approval of the ~~Plant~~ plant manager; and

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(continued)



5.5 Programs and Manuals

5.5.1 ODCM (continued)

- c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment Program

(169)

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include Containment Spray, Safety Injection, and Residual Heat Removal (RHR) in the recirculation configuration. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling Program

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for sampling and analysis; and
- c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program (continued)

- (169)
- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 112, Column 1;
 - h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
 - i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
 - j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

(156) 5.5.5 Component Cyclic or Transient Limit Program

(200) This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in the UFSAR Table 5.1.4 to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

(240) This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with a NRC approved program.

Regulatory Guide 1.35, Revision 2
The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)



5.5 Programs and Manuals (continued)

5.5.7 ~~Reactor Coolant Pump Flywheel Inspection~~ Inservice Testing Program

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~~This program shall provide~~ provides controls for the ~~inspection~~ inservice testing of each reactor coolant pump flywheel ~~per the recommendations of Regulation Position~~ ASME Code Class 1, 2, and 3 components including applicable supports. ~~4-b of Regulatory Guide 1.14, Revision 1, August 1975.~~

5.5.8 ~~Inservice Testing Program~~

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~~This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports, high energy piping outside of containment, and steam generator tubes.~~ The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities

Required Frequencies for performing inservice testing activities

Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

(continued)

5.5 Programs and Manuals (continued)

- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

(continued)

in accordance with the Nuclear Policy Manual.
This inspection program shall define the specific
requirements of the edition and Addenda of the
ASME Boiler and Pressure Code, Section XI, as
required by 10 CFR 50.55a(g)

5.5 Programs and Manuals (continued)

5.5.95 5.8 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program. The program shall include the following:

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- a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program.
 - b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.
 - c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.10 ~~Ventilation Filter Testing (VFTP)~~ 5.5.9 Secondary Water Chemistry Program

A

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~~This program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems to inhibit SG tube degradation. The test frequencies and methods, where practical, will be performed in accordance with Regulatory Guide 1.52.~~

~~a. Post Accident Charcoal System~~

- ~~1. Demonstrate the pressure drop across the charcoal absorber bank is < 3 inches of water at a design flow rate ($\pm 10\%$).~~
- ~~2. Demonstrate that an in-place Freon test of the charcoal absorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.~~
- ~~3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.~~

~~b. Containment Recirculation System~~

- ~~1. This program shall include:~~

(continued)

5.5 Programs and Manuals (continued)

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

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(continued)

5.5 Programs and Manuals (continued)

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies and methods will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented.

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a. Containment Post-Accident Charcoal System

1. Demonstrate the pressure drop across the charcoal adsorber adsorber bank is < 3 inches of water at a design flow rate ($\pm 10\%$).

2. Demonstrate that an in-place Freon test of the charcoal adsorber adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.

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b. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate ($\pm 10\%$).

5.5.10 VFTP (continued)

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

c. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the HEPA filter bank is < 3 inches of water at a design flow rate ($\pm 10\%$).

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2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 VFTP (continued)

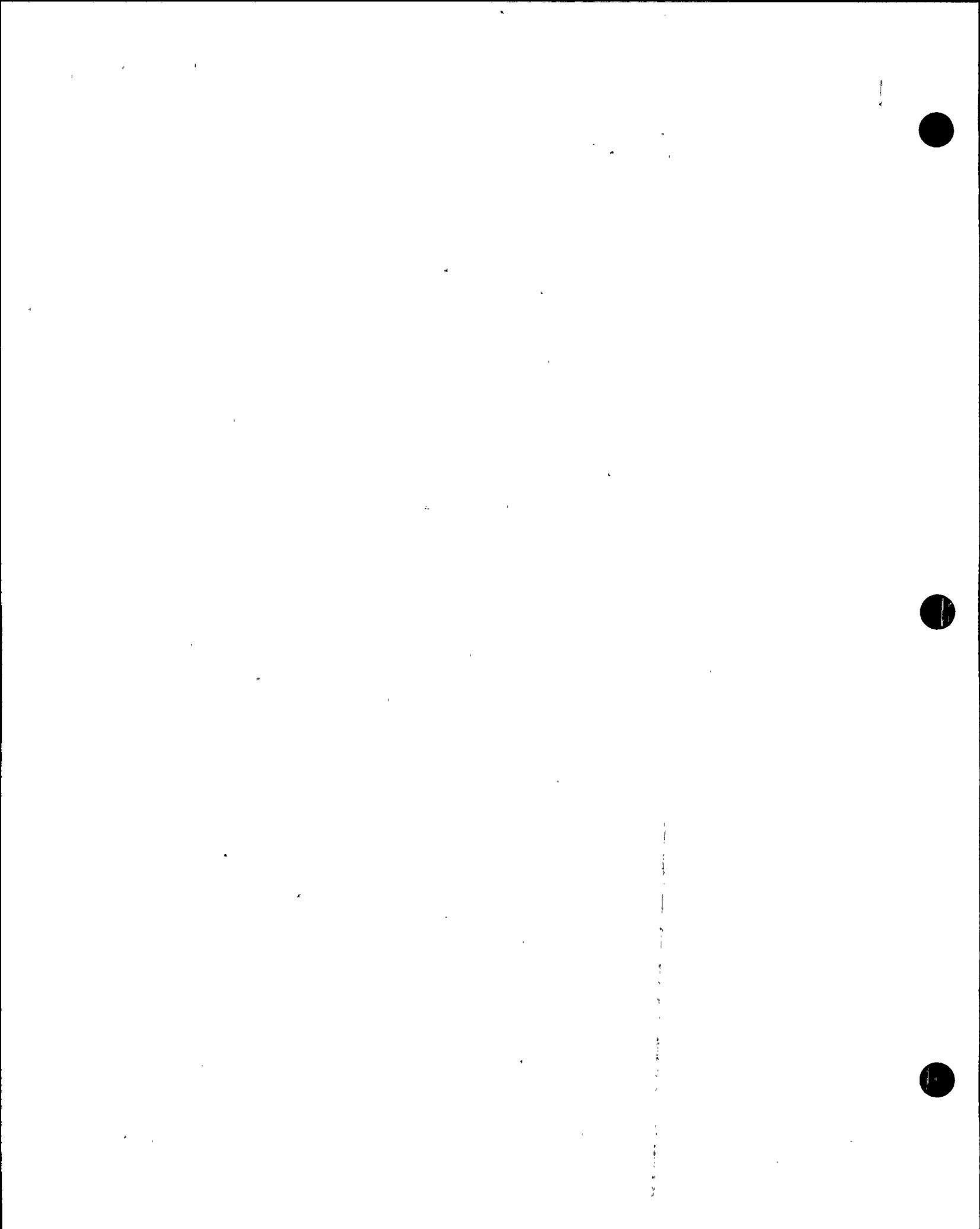
3. Demonstrate the pressure drop across the charcoal ~~adsorber~~ adsorber bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
4. Demonstrate that an in-place Freon test of the charcoal ~~adsorber~~ adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.
5. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.

d. Spent Fuel Pit SFP Charcoal Absorber ~~Adsorber~~ System

1. Demonstrate that the total air flow rate from the charcoal ~~absorbers~~ desorbers shows at least 75% of that measured with a complete set of new ~~absorbers~~ desorbers.
2. Demonstrate that an in-place Freon test of the charcoal ~~adsorbers~~ adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.
3. Demonstrate for a carbon sample that a laboratory analysis shows the iodine removal efficiency of $\geq 90\%$ of radioactive methyl iodide.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

(continued)



5.5 Programs and Manuals (continued)

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NUREG-0133.

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and
- b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ~~limits specified in Table 1 of applicable~~ ASTM D975 ~~Standards~~.

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(continued)



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5.5 Programs and Manuals (continued)

The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
1. an API gravity or an absolute specific gravity within limits,
 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Diesel Fuel Oil Testing Program (continued)

3. a clear and bright appearance with proper color; and

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b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 20 fuel oil.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. A change in the TS incorporated in the license; or
 2. A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71e.

(continued)



5.5 Programs and Manuals (continued)

5.5.14 Safety Function Determination Program (SFDP)

This program ensures that a loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. ~~These appropriate actions will be stated in an exception to entering supported system Condition and Required Actions as provided in the Specifications.~~ This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

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- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
 - b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
 - c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
 - d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system ~~or train is~~ inoperable ~~(i, and:~~

- 187
- a. ~~eA required system redundant to the supported system(s) is also inoperable; or~~
 - b. ~~, the Conditions and Required Actions have been entered), and:~~
 - a. ~~A required system or train redundant to the supported system(s) or train is also inoperable; or~~
 - b. ~~A required system redundant to the system(s) or train supported by the inoperable supported system or train is also inoperable; or~~

(continued)

5.5 Programs and Manuals (continued)

(continued)

5.5 Programs and Manuals (continued)

5.5.14 ~~SFDP (continued)~~

~~c. A required system or trains redundant to the inoperable support system(s) or train for the supported systems (a) and (b) above is also inoperable.~~

(187) ~~The SFDP identifies where a loss of safety function exists. A required system or train redundant to the supported system(s) or train in turn supported by the inoperable supported system is also inoperable; or~~

~~b.c. A required system redundant to the inoperable support system(s) or train for the supported by the inoperable supported system or trains systems (a) and (b) above is also inoperable; or~~

(continued)

5.5 Programs and Manuals (continued)

5.5.14 ~~SFDP~~ (continued)

(187)

e.

(continued)

5.5 Programs and Manuals

~~A required system or trains redundant to the inoperable support system(s) or train for the supported systems (a) and (b) above is also inoperable.~~ 5.5.14
SFDP (continued)

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The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 ~~Secondary Water Chemistry Program~~

~~This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:~~

- ~~a. Identification of a sampling schedule for the critical variables and control points for these variables;~~
- ~~b. Identification of the procedures used to measure the values of the critical variables;~~
- ~~c. Identification of process sampling points;~~
- ~~d. Procedures for the recording and management of data;~~
- ~~e. Procedures defining corrective actions for all off control point chemistry conditions; and~~
- ~~f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate~~

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~~corrective action.~~ 5.5.15 ~~Containment Leakage Rate Testing Program~~

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

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The peak calculated containment internal pressure for the design basis loss of coolant accident, P_0 , is 60 psig.

The maximum allowable primary containment leakage rate, L_0 , at P_0 , shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

5.5 Programs and Manuals

a. Containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) For each air lock, overall leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$, and

2) For each door, leakage rate is $\leq 0.01 L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05 L_a$ when tested at $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.



5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

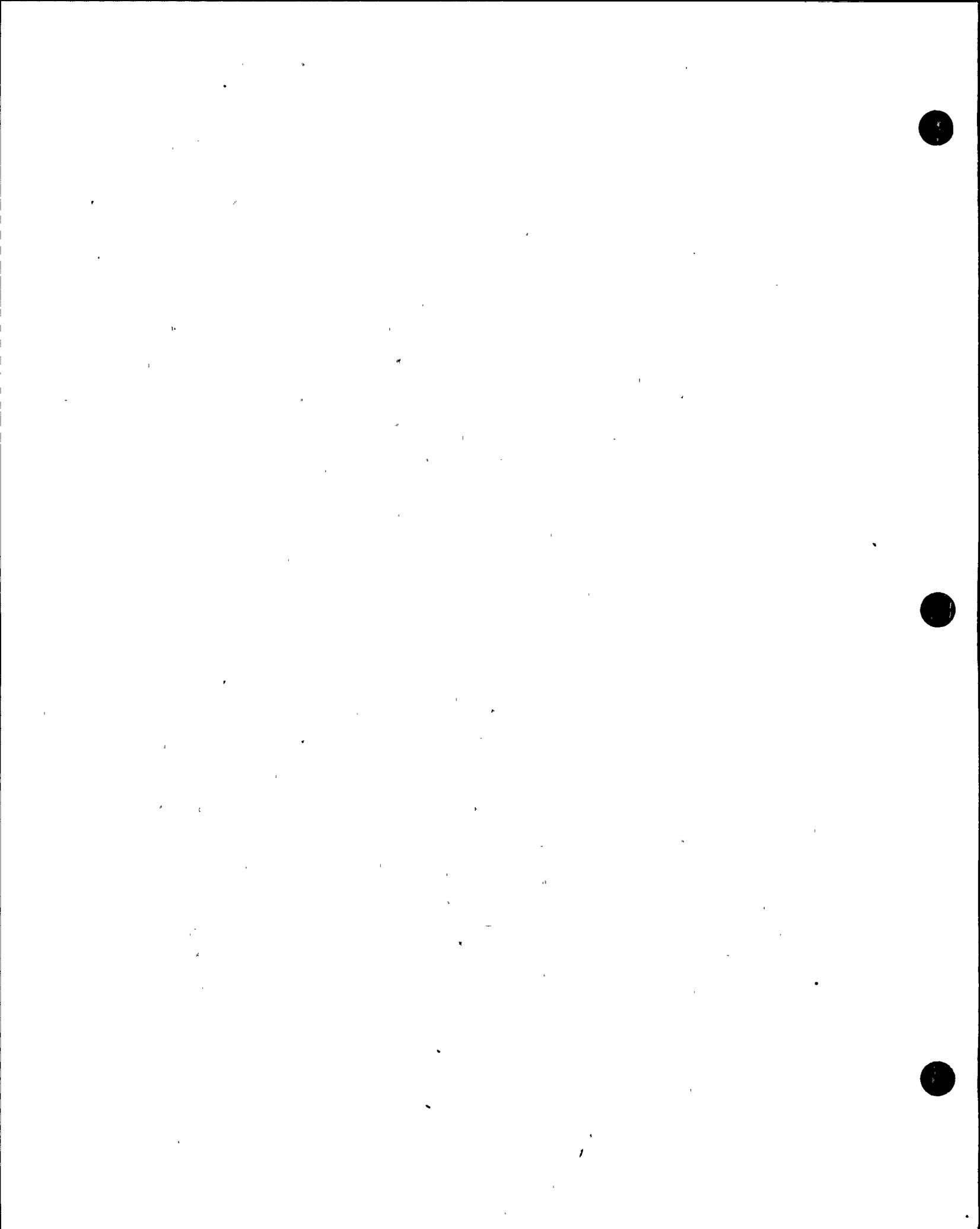
A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted on or before April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)



5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the plant shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
 LCO 3.1.3, "MODERATOR TEMPERATURE COEFFICIENT (MTC)";
 LCO 3.1.5, "Shutdown Bank Insertion Limit";
 LCO 3.1.6, "Control Bank Insertion Limits";
 LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
 LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)";
 LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
220 ~~LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation";~~
~~LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits";~~
~~LCO 3.5.1, "Accumulators";~~
~~LCO 3.5.4, "Refueling Water Storage Tank (RWST)";~~
~~LCO 3.7.12, "Spent Fuel Pool (SFP) Boron Concentration";~~ and
 LCO 3.9.1, "Boron Concentration";

(continued)

5.6 Reporting Requirements (continued)

5.6.5 ~~COLR~~ (continued)

~~Concentration.~~

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(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, (~~"Westinghouse~~ Westinghouse Reload Safety Evaluation Methodology," July 1985.
+
(Methodology for LCO 3.1.1, LCO 3.1.3, LCO 3.1.5, ~~LCO 3.1.6, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, LCO 3.5.1, LCO 3.5.4 and LCO 3.9.1.)~~)
2. WCAP-9220-P-A (~~"Westinghouse,~~ Westinghouse ECCS Evaluation Model-1981 Version," Revision 1, February 1982.
+
(Methodology for LCO 3.2.1.)
3. WCAP-8385, (~~"Power~~ Power Distribution Control and Load Following Procedures - Topical Report," September 1974.
+
(Methodology for LCO 3.2.3.)
4. ~~WCAP-8745-P-A ("Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions~~ WCAP-8567-P-A, "Improved Thermal Design Procedure," ~~September 1986~~ February 1989.
+
(Methodology for LCO ~~3.3.1, Overtemperature ΔT and Overpower ΔT trip setpoints~~)

~~5.3.4.1 when using ITDP.~~

~~WCAP-8567-P-A, ("Improved Thermal Design Procedure," February 1989)~~

~~5.3.4.1 when using RTDP.~~
+
~~(Methodology for LCO 3.4.1 when using ITDP~~ WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.
~~(Methodology for LCO 3.4.1 when using RTDP.)~~

6. ~~WCAP-11397-P-A, ("Revised Thermal Design Procedure~~ WCAP-10054-P-A and WCAP-10081, "Westinghouse Small Break

(continued)



5.6 Reporting Requirements

~~ECCS Evaluation Model Using the NOTRUMP Code," April 1989~~
~~August 1985.~~

+
(Methodology for LCO 3.4.1 when using RTDP3 2.1)

7. +

~~7. WCAP 11596 P A, ("Qualification of the PHOENIX P/ANG Nuclear Design System for Pressurized Water Reactor Cores," June 1988.)~~

~~(Methodology for LCO 3.7.12.)~~

(continued)



5.6 Reporting Requirements

5.6.5 ~~COLR~~ (continued)

~~e.WCAP 11596 P AWCAP 10924 P A, ("Qualification of the PHOENIX-
P/ANC Nuclear Design System for Pressurized Water Reactor Cores,"
June 1988 Volume 1, Rev. 1)~~

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~~(Methodology for LCO 3.7.121, and Addenda 1, 2, 3;
Westinghouse Large Break LOCA Best-Estimate
Methodology, Volume 1: Model Description and
Validation, December 1988.)~~

(continued)

5.6 Reporting Requirements

5.6.5 ~~COLR~~ (continued)

e(Methodology for LCO 3.2.1)

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8. WCAP-10924-P-A, Volume 2, Rev. 2, and Addenda, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," December 1988. (Methodology for LCO 3.2.1)

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

9. WCAP-10924-P-A, Rev. 2 and WCAP-12071, "Westinghouse Large-Break LOCA Best Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped With Upper Plenum Injection, Addendum 1: Responses to NRC Questions," December 1988.
(Methodology for LCO 3.2.1)

10. WCAP-10924-P, Volume 1, Rev. 1, Addendum 4, "Westinghouse LBLOCA Best Estimate Methodology; Model Description and Validation; Model Revisions," August 1990.
(Methodology for LCO 3.2.1)

- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heatup, cooldown, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
- LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
- b. The power operated relief valve lift settings required to support the Low Temperature Overpressure Protection (LTOP) System, and the LTOP enable temperature shall be established and documented in the PTLR for the following:
- LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves"; and
LCO 3.4.12, "LTOP System."

(continued)

5.6 Reporting Requirements

⁶
~~5.6.5~~ ~~PTLR (continued)~~ and LTOP

(205)
c. The analytical methods used to determine ~~changes to~~ the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically, ~~those~~ ^{the limits and methodology are} described in the following documents:

(220)
³⁻¹⁸ ~~WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994~~ ~~Letter from C.~~

in Amendment No. 48. The acceptability of the PIT and LTOP limits are documented in NRC letter "R.E. Ginna - Acceptance for Referencing of Pressure Limit Report," December 1995.

(continued)

5.6 Reporting Requirements

~~5.6.6 PTLR (continued)~~

21. Grimes, NRC, to R. A. Newton, WOG, Subject: "Acceptance for Referencing Topical Report WCAP-14040, Revision 1, Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves (TAC #M91749)," dated October 16, 1995.
(Methodology for LCO 3.4.3).

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22. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: "Technical Specification Improvement Program, RCS Pressure and Temperature Limits Report," dated May/December 5, 1995.

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(Methodology for LCOs 3.4.6, 3.4.7, 3.4.10 and 3.4.12 LTOP Enable Temperature and LCO 3.4.12 - Pressurizer Power Operated Relief Valve Lift Setting Limits.)

d. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for ~~any~~ revision or supplement thereto.

1. Amendment No 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant, dated March 6, 1992.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(a), in lieu of the requirements of 10 CFR 20.1601(c), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr at a distance of 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., ~~Radiation Protection Technicians~~ ~~radiation protection technicians~~) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.
- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~radiation protection technician~~ in the RWP.

(continued)

5.7 High Radiation Area (continued)

- 5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels > 1000 mrem/hr at a distance of 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.
- 5.7.3 In addition to the requirements of Specification 5.7.1, for individual high radiation areas with radiation levels of > 1000 mrem/hr at a distance of 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.
-

TABLE 1

RELOCATED TO CORE OPERATING LIMITS REPORT (COLR)
 (Changes controlled by Specification 5.6.5 and 10 CFR 50.59)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
20.iii	3.10.1.1, Figure 3.10-2	Shutdown Margin Limits
20.v	3.10.1.3, Figure 3-10.1	Control Bank Limits
20.xxiii	3.10.2.2	$F_Q(Z)$ and F_{AH}^N Limits
20.xxxv	3.10.2.10a	AFD Target Band
20.xxxiii	3.10.2.8	AFD Target Band

(1) From Attachment A, Section D

TABLE 2

RELOCATED TO UFSAR

(Changes controlled by: (1) 10 CFR 50.54 or (2) 10 CFR 50.59)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION	CHANGE CONTROL
15.i.a	Table 3.5-1	Various Instrumentation Operational Details	(1)
15.ii.a	Table 3.5-2	Various Instrumentation Operational Details	(1)
15.ii.r	Table 3.5-2, Notes	ESFAS Instrumentation Design	(2)
15.ii.r	Table 3.5-2, Notes	ESFAS Instrumentation Design	(2)
15.iii.a	Table 3.5-3	Various Instrumentation Operational Details	(1)
44.i	5.1	Site	(2)
45.i	5.2	Containment Design Features	(2)
46.i	5.3.1.a & 5.3.1.c	Miscellaneous Reactor Core Design Features	(2)
46.iii	5.3.2	RCS Design Features	(2)
47.iii	5.4.3	SFP Region I Decay Time Limit	(2)
48.i	5.5	Waste Treatment Systems	(2)
50.ii	6.1.1 & 6.2.1	Management Titles	(1)(2)
52.i	6.4	Training	(1)(2)
57.ii	6.9.1.1	Startup Report	(2)

(1) From Attachment A, Section D

TABLE 3

RELOCATED TO IST PROGRAM
 (Changes controlled by Specification 5.5.8 and 10 CFR 50.55a(f))

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
28.ii.c	Table 4.1-2, #7	Pressurizer Safety Valve Testing Frequency
28.ii.m	Table 4.1-2, #12	Fire Protection Pump Information
29.i	4.2	ISI/IST Program Information
32.ii	4.5.2.1	CS, SI & RHR Pump Head Limits
32.iii	4.5.2.2.c	Accumulator Check Valve Testing
35.i	4.8.1 & 4.8.2	AFW Pump Tests
35.ii	4.8.3	AFW Valve Tests
35.iii	4.8.4	SAFW Pump Tests
35.v	4.8.6	AFW & SAFW Tests
38.v	4.11.2.2	RHR Pump Testing in MODE 6

(1) From Attachment A, Section D

TABLE 4

RELOCATED TO ISI PROGRAM
(Changes controlled by 10 CFR 50.55a(g))

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
23.i	3.13	Snubbers
29.i	4.2.1	ISI/IST Program Information
41.i	4.14	Snubber Surveillance Program

(1) From Attachment A, Section D

TABLE 5

RELOCATED TO ODCM
(Changes Controlled by Specification 5.5.1)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
15.iv	3.5.4 & Table 3.5-6	Radiation Accident Monitoring
15.viii	3.5.5 & Table 3.5-5	Radioactive Effluent Monitoring Instrumentation
19.i	3.9.1.1	Liquid Effluents Concentration
19.ii	3.9.1.2 & 3.9.2.4	Liquid Effluents Dose
19.iii	3.9.1.3	Liquid Waste Treatment
19.iv	3.9.2.1	Gaseous Wastes Dose Rate
19.v	3.9.2.2.a, 3.9.2.2.c, 3.9.2.4	Gaseous Wastes Dose
19.vi	3.9.2.2.b, 3.9.2.2.c, 3.9.2.4	Gaseous Waste Dose
19.vii	3.9.2.3	Gaseous Waste Treatment
19.ix	3.9.2.7	Solid Radioactive Waste
26.i	3.16.1 & Table 3.16-1	Radiological Environmental Monitoring Program
26.ii	3.16.2	Land Use Census
26.iii	3.16.3	Interlaboratory Comparison Program
28.i.j	Tbl. 4.1-1, #18, 28, 29	Radiation Monitoring Instrumentation
28.v.b	4.1.4 & Table 4.1-5	Radioactive Effluent Monitoring Surveillance Requirements
37.i	4.10.1 & Table 4.10-1	Radiological Environmental Monitoring
37.ii	4.10.2	Land Use Census
37.iii	4.10.3	Interlaboratory Comparison Program
39.i	4.12.1.1 & Table 4.12-1	Liquid Effluents Concentration
39.ii	4.12.1.2	Liquid Effluents Dose
39.iii	4.12.2.1 & Table 4.12-2	Gaseous Wastes Release Rate

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
39.iv	4.12.2.2	Gaseous Wastes Dose
39.v	4.12.3	Waste Decay Tanks
40.i	4.13	Radioactive Material Source Leakage Tests
57.iv	6.9.1.3, 6.9.1.4, Table 6.9-1, & Table 6.9-2	Administrative Control Reports

(1) From Attachment A, Section D



TABLE 6

RELOCATED TO PLANT PROCEDURES
(Changes controlled by 10 CFR 50.59)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
18.ii	3.8.1.b	Radiation Levels in Containment
18.v	3.8.1.f	Control Room and Manipulator Crane Communications
18.vii	3.8.1.c	Containment Audible Flux Monitor
28.ii.n	Table 4.1-2, #18	Secondary Coolant Samples
28.v.e	Table 4.1.5, Note 5	CHANNEL CALIBRATION and National Bureau of Standards
31.i	4.4.4	Containment Tendon Surveillance
31.ii	4.4.3	Recirculation Heat Removal Systems
32.v	4.5.2.3	Air Filtration Systems
33.x	4.6.2.c	Battery Test Trending
38.i, 38.ii	4.11.1.1	Spent Fuel Pool Charcoal Adsorber System
46.i	5.3.1.a	Reporting Requirements for Rod Filler Material
57.iii	6.9.1.2	Monthly Operating Report Database
57.vi	6.9.1.5	PORV/Safety Valve Challenges
65.i	6.17	Major changes to Radioactive Waste Treatment Systems
57.vii	6.9.2.1	Sealed Source Reporting Requirements

(1) From Attachment A, Section D

TABLE 7

RELOCATED TO PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
 (Changes Controlled by Specification 5.6.6)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
6.v	3.1.1.1.k	LTOP Enable Temperature
7.i	3.1.2.1.a, Figure 3.1.1, & Figure 3.1-2	RCS Heatup and Cooldown Curves
7.v	3.1.2.3	Pressurizer Heatup and Cooldown Rates
25.ii	3.15.1	LTOP Setpoints

(1) From Attachment A, Section D

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TABLE 8

RELOCATED TO TECHNICAL REQUIREMENTS MANUAL (TRM)
(Changes Controlled by 10 CFR 50.59)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
6.xi	3.1.1.6	Reactor Vessel Head Vents
11.i	3.1.6	RCS Chemistry
12.ii	3.2.1 & 3.2.1.1	CVCS in MODES 5 and 6
12.iii	3.2.2 & 3.2.4	CVCS above MODE 5
12.iv	3.2.3 & Table 3.2-1	CVCS Boron Concentration
15.i.q	Table 3.5-1, #17	Circulating Water Flood Protection
15.v	3.5.6.1	Control Room Toxic Gas Monitors
17.iii	3.7.2.1.b.2, 3.7.2.2.a, & 3.7.2.2.b	Second Offsite Power Source
20.xv	3.10.4.3.2.b.iii & Table 3.10-1	Misaligned Rod Accident Analysis Evaluation
21.ii	3.11.2	Fuel Movement Requirements in Aux Bldg
21.iii	3.11.3, 3.11.5	Fuel Movement Requirements in Aux Bldg
21.iv	3.11.4	Fuel Movement Requirements in Aux Bldg
22.i	3.12.1	Moveable Incore Instrumentation
28.i.g	Table 4.1-1, #34 & 35	Control Room Toxic Gas Monitors
28.i.m	Table 4.1-1, #14, 16, & 19	CVCS Surveillances
28.ii.d	Tbl. 4.1-2, #10	Fuel Movement Requirements in Aux Bldg
28.ii.h	Tbl. 4.1-2, #19	Circulating Water Flood Protection
28.ii.l	Tbl. 4.1-2, #4	CVCS Surveillances
30.i	4.3.5.6	Reactor Vessel Head Vent

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
33.vi	4.6.1.e.3(b)	DG Sequence Time Limits
47.iii	5.4.3	SFP Tornado Related Requirements
55.ii	6.7.1.b	Safety Limit Violation Response
55.iii	6.7.1.c	Safety Limit Violation Response
55.iv	6.7.1.d	Safety Limit Violation Response

(1) From Attachment A, Section D

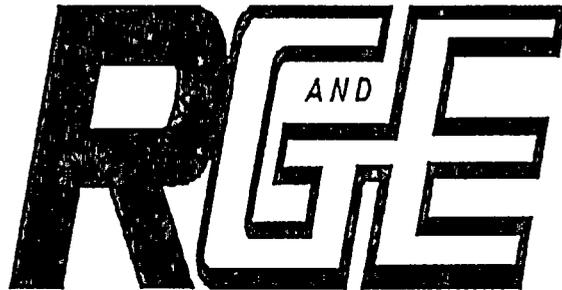
TABLE 9

RELOCATED TO TECHNICAL SPECIFICATION BASES
 (Changes controlled by Specification 5.5.13)

ITEM # ⁽¹⁾	CTS #	DESCRIPTION
13.ix	3.3.1.1.h	PIV Listing
13.xii	3.3.1.7	SI Pump Listing
13.xii	3.3.1.8	SI Pump Listing
16.v	3.6.5	Mini-purge valves opening restrictions*
32.viii	4.6.2.f	Battery degradation definition

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11/15/2014



Rochester Gas & Electric Corporation
R. E. Ginna Nuclear Power Plant

Improved Technical Specifications

December 1995 Submittal

Cover Letter and
Attachments A and B

Volume I

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PDR ADOCK 05000244
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D. JUSTIFICATION (CURRENT GINNA TS)

Converting to the ITS format will provide a significant human factors improvement by locating similar requirements within the same section and also provide a standard structure. In addition, the expanded bases information will support preparation of safety evaluations and training activities. There are several types of changes that are being requested by this LAR in order to perform the conversion. These changes are with respect to both the ITS and the current Ginna Station Technical Specifications. The technical and significant administrative changes related to the current Ginna Station TS are organized into multiple categories as summarized below.

i. Relocation of Requirements Within Technical Specifications

Many current specifications are moved to support consolidation of similar requirements within the same section. Since the requirements are only being relocated within the technical specifications, there is no reduction in safety. This category is mainly used to identify multiple requirements that are consolidated into a single new specification and not for listing requirements which are only renumbered.

ii. Elimination of Duplicated Regulatory Requirements

Several specifications currently duplicate existing regulatory requirements. The removal of these specifications eliminates the need to change technical specifications when there are rule changes. Since all licensees must meet the applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions, there are sufficient regulatory controls in place to allow elimination of duplicated requirements from technical specifications. The implementation of these requirements are contained in procedures and other licensee controlled documents.

iii. Relocation of Current Requirements To Other Controlled Documents

The relocation of certain requirements to other licensee controlled documents (i.e., UFSAR, QA Program, and plant procedures) does not eliminate the requirement. Instead, the requirements are relocated to other more appropriate documents and programs which have sufficient controls in place to manage implementation and future changes (e.g., 10 CFR 50.54(a)(3) and 10 CFR 50.59). The relocation of these items will enable RG&E to more efficiently maintain the requirements under existing regulations and reduce the need to request technical specification changes for issues which do not affect public safety.

iv. Addition of New ITS Requirements

There are several requirements contained in NUREG-1431 which are not currently in the Ginna Station Technical Specifications. These ITS requirements were added in order to provide a more complete specification. Changes within this category are further identified as either being a "more restrictive change" (iv.a) or a "less restrictive change" (iv.b).

v. Other Changes to Technical Specifications (Technical)

Several changes to existing requirements were made to provide consistency with NUREG-1431. Examples include moving requirements to LCO Notes and revising the current specified Completion Time. Also included within this category are the revision of the existing bases to reflect more current information. Changes within this category are further identified as either being a "more restrictive change" (v.a), "less restrictive change" (v.b), or an "administrative change" (v.c).

vi. Other Changes to Technical Specifications (Administrative)

Several minor changes to the technical specifications were made that are minor revisions only and do not involve any technical issues. Examples include updates of references to the Code of Federal Regulations.

The following section discusses changes to the current Ginna Station Technical Specifications which were not addressed in Section C of this attachment. This section is organized based on the existing TS chapter numbers to facilitate easier review. Each change is also identified with respect to one of the above categories (e.g., Ginna Station TS Category (i)). A marked up copy of the Ginna Station Technical Specifications is provided in Attachment B which identifies major changes only. A cross reference is provided in the margin of each specification that has been changed by use of a circle containing section numbers from below. For example, "1.i" found in the margin of the markup would refer to section 1.i below. A cross reference between the ITS and current Ginna Station Technical Specifications is also provided in Attachment E.

1. Technical Specification 1.0

- i. TS 1.2 - The definitions of operating MODES were revised as follows (these are Ginna TS Category (v.a) changes):
 - a. Refueling - see Note 1.ii below.
 - b. Cold Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits.

- c. Hot Shutdown - The reactivity limit was revised from $\leq -1 \Delta k/k\%$ to $< 0.99 k_{eff}$ which are equivalent limits. The average reactor coolant temperature was also revised from $\geq 540^\circ\text{F}$ to $\geq 350^\circ\text{F}$. This change eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2. The expansion of this temperature range is conservative since the current TS only use the Hot Shutdown MODE in two aspects. The first method is requiring a shutdown to this mode due to plant conditions. Since the upper temperature range for Hot Shutdown remains the same (i.e., the Operating MODE temperature), there is no impact. The second method is to require certain equipment to be OPERABLE in this mode. However, lowering the temperature limit to 350°F requires that the equipment would be OPERABLE for a larger temperature range.
- d. Operating - The reactivity limit was revised from $> -1 \Delta k/k\%$ to $\geq 0.99 k_{eff}$ which are equivalent limits. The average reactor coolant temperature of $\sim 580^\circ\text{F}$ was not added since this parameter is specified in new LCO 3.4.1. In addition, the Operating MODE was separated into two modes: Operating and Startup. The only difference between these two modes is that Startup is defined when the reactor is $\leq 5\%$ Rated Thermal Power (RTP) while the Operating MODE is when the reactor is $> 5\%$ RTP.
- e. A new operating mode (Hot Standby) was provided between Hot Shutdown and Cold Shutdown. This mode is defined as when the reactivity condition is $< 0.99 k_{eff}$ and the average reactor coolant temperature is $< 350^\circ\text{F}$ and $> 200^\circ\text{F}$ when the reactor vessel head closure bolts are fully tensioned. The definition of this new mode eliminates the use of an intermediate mode of 350°F as found throughout the current TS which is not defined in TS 1.2.

- ii. TS 1.3 - This definition of refueling was deleted. The current TS 1.2 provides a definition of refueling as being the reactor mode when reactivity is $\leq -5 \Delta k/k\%$ and the average reactor coolant temperature is $\leq 140^\circ\text{F}$. TS 1.3 states that refueling is "any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted" which is a subset of the mode defined in TS 1.2. The new TS Table 1.1-1 states that refueling is any condition in which "one or more reactor vessel head closure bolt is less than fully tensioned" with fuel in the reactor. While an average reactor coolant temperature or reactivity limit is no longer provided for the refueling mode definition, the reactor vessel head closure bolts cannot be removed at elevated reactor coolant temperatures or when the RCS is pressurized due to their design. A reactivity limit is also not required when the RCS is depressurized. Therefore, the new definition of the refueling mode is more conservative than current TS 1.3 and generally consistent with TS 1.2. This is a Ginna TS Category (v.a) change.
- iii. TS 1.5 - The definition for Operating was not added to the new specifications since it is no longer required. This definition is addressed by the new definition for OPERABLE - OPERABILITY. This is a Ginna TS Category (i) change.
- iv. TS 1.6 - The definition for Degree of Redundancy (Instrument Channels) was not added to the new specifications since it is no longer required. This definition is addressed within new TS 3.3 (Instrumentation). This is a Ginna TS Category (v.c) change.
- v. TS 1.7.1 - This was revised to specify that the CHANNEL CALIBRATION includes the required interlock and time constant functions of the channel. In addition, discussion of calibrating instrument channels with resistance temperature detectors was added for clarification. These are Ginna TS Category (v.a) changes.
- vi. TS 1.7.2 - The last sentence of this definition was revised as follows:

This determination shall include, where possible, comparison of the channel indication and/or status with to other indications and/or status derived from independent instrumentation channels measuring the same parameter.

These minor changes provide greater clarification of the defined term and are Ginna TS Category (v.c) changes.

- vii. TS 1.7.3 - The definitions for testing of analog and bistable channels were combined into one description with a new title. The only difference between the two definitions is that testing of bistable channels required injection of a simulated or source signal into the sensor versus "as close to the sensor as possible" for analog channels. Since the bistable must be actuated to determine operability, maintaining the analog channel description for the combined definition is acceptable. In addition, the combined definition was expanded to require "adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy." These are Ginna TS Category (v.a) changes.
- viii. TS 1.7.4 - The definition for Source Check was not added to the new specifications since it is no longer required. The performance of a Source Check is now addressed within the definition of CHANNEL CALIBRATION and CHANNEL OPERATING TEST (COT). This is a Ginna TS Category (v.c) change.
- ix. TS 1.8 - The definition for Containment Integrity was relocated to the bases of new TS 3.6.1 and 3.6.2 which essentially requires compliance with 10 CFR 50, Appendix J and the GDC. This is a Ginna TS Category (iii) change.
- x. TS 1.10 - The definition for Hot Channel Factors was not added to the new specifications since it is no longer required. The Hot Channel Factor limit is only discussed in one LCO with the limit defined in the COLR. This is a Ginna TS Category (v.c) change.
- xi. TS 1.11 - This previously deleted definition was not added to the new specifications. This is a Ginna TS Category (vi) change.
- xii. TS 1.12 - The Frequency for Surveillance Requirements is now specified in hours, days or months in the new specifications such that the current definition of Frequency Notation is no longer required. Consequently, this definition was replaced with a general description of how to use and apply the Frequency requirements. In addition, the definition of refueling Frequency was revised from 18 months to 24 months for all systems. This is discussed in Attachment H and is a Ginna TS Category (v.b.1) change.
- xiii. TS 1.13 - The definition for Offsite Dose Calculation Manual (ODCM) was moved to the ODCM program description in ITS specification 5.5.1. The change to the CTS is editorial because the program description involves reorganization or reformatting of requirements without affecting technical content. This is a Ginna TS Category (v.c) change.

- xiv. TS 1.14 - The definition for Process Control Program (PCP) was not added to the new specifications since it is no longer required. The PCP was relocated from the technical specifications to the TRM and does not need to be described within new TS 1.1. This is a Ginna TS Category (v.c) change.
- xv. TS 1.15 - The definition for Solidification was not added to the new specifications since it is no longer required. Solidification is described within the PCP which was relocated from the technical specifications to the TRM. Therefore, this definition does not need to be provided in new TS 1.1. This is a Ginna TS Category (v.c) change.
- xvi. TS 1.16 - The definition for Purge - Purging was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3. This is a Ginna TS Category (v.c) change.
- xvii. TS 1.17 - The definition for Venting was not added to the new specifications since it is no longer required. This definition only pertains to the Containment Purge system which is described in new TS 3.6.3. This is a Ginna TS Category (v.c) change.
- xviii. Not used.
- xix. TS 1.19 - The definition for Reportable Event was not added to the new specifications since it is no longer required. Reportable Events are described in 10 CFR 50.72 and 50.73. This is a Ginna TS Category (ii) change.
- xx. TS 1.20 - The definition for Canisters Containing Consolidated Fuel Rods was not added to the new specifications since it is no longer required. This definition is provided in new TS 4.3 which is the only section that addresses consolidated fuel rods. This is a Ginna TS Category (v.c) change.

- xxi. TS 1.21 - The definition for Shutdown Margin was expanded to require another assumption that in MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature. Also, the definition was revised to require consideration of any RCCA known to be incapable of being fully inserted. This is in addition to the existing assumptions related to a stuck fully withdrawn single RCCA with the highest reactivity worth. The definition description discussing "no changes in xenon or boron concentration" was deleted since this level of detail is not required. These clarifications, which are consistent with NUREG-1431, are Ginna TS Category (v.a) changes.
- xxii. TS 1.4 - The definition for OPERABLE -. OPERABILITY was revised to remove "supports." This phrase was added to the current definition by Reference 3 but is not consistent with the definition as provided in NUREG-1431. Therefore, to provided consistency, this was not added to the new specifications. This is a Ginna TS Category (v.c) change.
- xxiii. The following definitions were added to the new specifications since the associated terms are used throughout the document (these are Ginna TS Category (v.a) changes):
- a. ACTIONS
 - b. ACTUATION LOGIC TEST
 - c. AXIAL FLUX DIFFERENCE
 - d. CORE ALTERATION
 - e. CORE OPERATING LIMITS REPORT (COLR)
 - f. LEAKAGE
 - g. PHYSICS TESTS
 - h. PRESSURE TEMPERATURE LIMITS REPORT (PTLR)
 - i. RATED THERMAL POWER
 - j. STAGGERED TEST BASIS
 - k. TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)
- xxiv. A new section was added to the specifications which explains the use of Logical Connectors within the new TS. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.
- xxv. A new section was added to the specifications which explains the use of the Completion Time convention within the new TS. There are several changes from the current Ginna Station TS format which are discussed in this section (these are Ginna TS Category (v.a) changes):

- a. Completion Times in the new TS are based on the format that the clock for all Required Actions begin from the time that the Condition is entered. The Completion Times in the new specifications and the current Ginna Station TS are typically equal. For example, the new specifications may require that the plant be in MODE 3 within 6 hours and in MODE 4 within 36 hours for a specified Condition while the current Ginna Station TS require that the plant be in MODE 3 within 6 hours and in MODE 4 within an additional 30 hours for the same Condition. The intent of both the new specifications and the current Ginna Station TS is the same (i.e. be in MODE 4 within 36 hours).
- b. The new specifications restrict multiple entries into the ACTION table for separate Conditions unless it is specifically stated as acceptable. For example, if one SI pump is inoperable and during the LCO, a second SI pump is declared inoperable, the plant would enter 3.0 conditions in both the new specifications and the current Ginna Station TS. If the first SI pump were restored to OPERABLE status before entering MODE 3, the plant could resume operation in both TS. However, in the current TS, the Completion Time for restoring the second SI pump to OPERABLE status would begin from the time that it was declared inoperable. In the new specifications, the Completion Time would begin from the time the first pump was declared inoperable with an additional 24 hours allowed. This is a conservative change.

xxvi. A new section was added to the specifications which explains the use of the Frequencies specified within the SRs. This section does not provide any new requirements, only a description and examples of how to use the new ITS format. This is a Ginna TS Category (v.c) change.

2. Technical Specification 2.1

- i. The Applicability was revised to define when the reactor is in "operation" as MODES 1 and 2. This is an editorial change only since "operation" has been redefined as MODES 1 and 2 per Section D Change 1.i.d. This is a Ginna TS Category (iv.a) change.

3. Technical Specification 2.2

- i. The Applicability was revised to "MODES 1, 2, 3, 4, and 5." The proposed Applicability does not require this Safety Limit (SL) to be met when fuel is in the vessel with one or more reactor vessel head closure bolts less than fully tensioned or with the head removed. With the reactor head bolts less than fully tensioned, it is highly unlikely that the RCS can be pressurized greater than the SL pressure due to the low temperature over-pressure protection requirements. With the head removed, it is not possible to pressurize the RCS greater than the SL pressure. This is a Ginna TS Category (v.b.2) change.

4. Technical Specification 2.3

- i. This entire section was relocated to ITS Chapter 3.3, "Instrumentation." This is a Ginna TS Category (i) change.
- ii. TS 2.3 - Various limiting safety system settings (LSSS) are addressed as "Trip Setpoints," "Allowable Values," or "Applicable Modes" (as permissives) for their respective Reactor Trip System (RTS) instrumentation Functions in new LCO 3.3.1. Specific changes to the LSSS are discussed below for each of the associated Functional Units. This is a Ginna TS Category (i) change.
- iii. Not used.
- iv. TS 2.3.3.1, TS 2.3.3.2, and Figure 2.3-1 - The LSSS for the loss of voltage and degraded voltage functions were revised to provide a minimum Trip Setpoint value. Criteria for the establishment of equivalent values based on measured voltage versus relay operating time was relocated to the bases for LCO 3.3.4. This is a Ginna TS Category (iii) change.
- v. TS 2.3.2 - The listing of permissives was revised to provide requirements and setpoints for P-6, P-9, and P-10. These permissives also provide enabling and blocking features for various RTS functions. This is a Ginna TS Category (v.a) change.

5. Technical Specification 3.0

- i. A new section LCO 3.0.1 was added which explains the use of the Applicability statement in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.1. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- ii. A new section LCO 3.0.2 was added which explains the use of the associated ACTIONS upon discovery of a failure to meet an LCO in the new TS. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with LCO 3.0.2. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- iii. TS 3.0.1 - This was revised to clarify the use of the actions that must be implemented when an LCO is not met and (1) an associated Required Action and Completion Time is not met and no other Condition applies, or (2) the condition of the plant is not specifically addressed by the associated ACTIONS. The current requirement that the LCO time limits apply if they are more limiting than those required by LCO 3.0.3 is deleted and an expanded discussion is provided in the Basis to clarify the applicability of this requirement. This section does not provide any new requirements except as discussed in item 5.viii below. The clarifications and examples are based on the use of the new ITS format. This is a Ginna TS Category (v.c) change.
- iv. A new section LCO 3.0.4 was added which explains the limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met in the new TS. This section provides new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.

- v. A new section LCO 3.0.5 was added to provide an exception to LCO 3.0.2 for instances where restoration of inoperable equipment to an OPERABLE status could not be performed while continuing to comply with Required Actions. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as: maintaining an isolation valve closed or tripping an inoperable instrument channel. To allow the performance of SRs to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the present Specifications. Without this allowance certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications to preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing. Since this requirement is informally utilized and has no licensing basis, this section is considered to provide new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.
- vi. TS 3.0.2 - This was deleted and replaced by LCO 3.0.6 which provides guidance regarding the appropriate ACTIONS to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). Since its function is to clarify existing ambiguities and to maintain actions within the realm of previous industry interpretations and NRC positions, this new provision does not provide any new requirements. The information contained in TS 3.0.2 was relocated to LCO 3.8.1 which allows one power source to a safeguards bus and a redundant safety features on a second bus to be inoperable for 12 hours versus 1 hour. This change is consistent with NUREG-1431. These are Ginna TS Category (v.c) and (i) changes, respectively.

- vii. A new section LCO 3.0.7 was added to provide guidance regarding Test Exceptions for LCO 3.1.8. This LCO allows specified Technical Specification requirements to be changed (i.e., made applicable in part or whole, or suspended) to permit the performance of special tests or operations which otherwise could not be performed. If this Test Exception LCO did not exist, many of the special tests and operations necessary to demonstrate select plant performance characteristics, special maintenance activities and special evolutions could not be performed. This Specification eliminates the confusion which would otherwise exist as to which LCOs apply during the performance of a special test or operation. Without this specific allowance to change the requirements of another LCO, a conflict of requirements could be incorrectly interpreted to exist. This section does not provide any new requirements. This LCO provides clarifying and descriptive information for the LCOs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.
- viii. TS 3.0.1 - This was revised to remove the 1 hour allowance to prepare for a plant shutdown. Instead, the plant must now be in hot shutdown (i.e., MODE 3) within 6 hours of entering this LCO and cold shutdown (i.e., MODE 5) within 36 hours. No time limits are now placed on initiating the plant shutdown, only in the time frame in which the shutdown must be completed. Since the plant must now be in a lower mode in less amount of time, this is a more restrictive change. However, since no restrictions are made as to when the shutdown must commence, this is identified as a Ginna TS Category (v.c) change.

6. Technical Specification 3.1.1

- i. TS 3.1.1.1.b - This requirement was changed to require entry into MODE 1 \leq 8.5% RTP within six hours versus an immediate power reduction under administrative control. This change defines a specific number of hours to reach this condition which provides greater clarity to the operators. The remaining actions as specified by TS 3.1.1.1.b were relocated to LCO 3.4.5 and are discussed in 6.ii below. This is a Ginna TS Category (v.a) change.

- ii. TS 3.1.1.1.b, 3.1.1.1.c, and 3.1.1.1.d - These requirements were revised per new LCO 3.4.5 to require both reactor coolant loops OPERABLE with one loop in operation during MODES 1 \leq 8.5% RTP, and MODES 2 and 3, versus one in operation and the other OPERABLE for natural circulation between 350°F and 8.5% RTP. However, one RCS loop is now allowed to be inoperable for up to 72 hours provided that the shutdown margin as provided in the COLR is maintained and the non-operating RCS loop is OPERABLE (i.e., available for natural recirculation). These are all conservative changes (Ginna TS Category (iv.a) changes) since:
 - a. Two RCS loops are required to be OPERABLE.
 - b. A defined period of time is now specified for one RCS loop operation which addresses the concern raised by Reference 12. In addition, Completion Times are now specified for verifying shutdown margin and natural circulation capability.
- iii. TS 3.1.1.1.f - The exception for not requiring the RCS or RHR loops during steam generator crevice cleaning operations was not added to the new specifications since RG&E no longer performs this activity and the new SGs scheduled to be installed in 1996 do not have crevices subjected to cleaning as described in this specification. This is a conservative deletion and is a Ginna TS Category (v.a) change.
- iv. TS 3.1.1.1.g - The action to be in Cold Shutdown (i.e., < 200°F) within 24 hours was not added for the Condition with both RHR loops inoperable and only one RCS loop inoperable consistent with Condition B of LCO 3.4.6. Since RHR is the only system which provides long-term decay heat removal below 200°F, it is not prudent to bring the plant to a lower MODE until RHR is recovered. This is a Ginna TS Category (v.a) change.
- v. TS 3.1.1.1.k - This requirement was changed into a Note for LCO 3.4.6 and 3.4.7. This is a Ginna TS Category (v.c) change. The value for LTOP enable temperature with respect to the RCPs was also relocated to the PTLR. This is a Ginna TS Category (iii) change.

- vi. TS 3.1.1.1.f - This requirement was revised to require one RHR loop to be operating when in MODE 5 consistent with LCO 3.4.7 and 3.4.8. A RHR pump is required to be operating since a RCP cannot be routinely operated under these low temperature and pressure conditions. However, a SG with minimum water level of 16% can provide an alternate means of decay heat removal to the operating RHR loop in MODE 5 with the loops filled. In addition, a limit of 15 minutes (versus 1 hour) was placed on removing both RHR loops from service in MODE 5 with the loops not filled due to the reduced RCS inventory. These are conservative changes to the current requirements and are Ginna TS Category (v.a) changes.
- vii. TS 3.1.1.1.e - The note associated with the power sources for the RHR loops has been relocated to the specifications for electrical requirements during MODES 5 and 6 (i.e., LCOs 3.8.2, 3.8.5, 3.8.8, and 3.8.10). This is a Ginna TS Category (i) change.
- viii. TS 3.1.1.1.i and 3.1.1.1.j - These requirements were not added due to the expanded specifications provided in new TS 3.4.4, 3.4.5, 3.4.6, 3.4.7, and 3.4.8. The new specifications ensure that the appropriate RCS or RHR loop is available to provide forced flow for decay heat removal and boron mixing. Therefore, these requirements are no longer necessary. This is a Ginna TS Category (v.c) change.
- ix. TS 3.1.1.5.a - The lower limit for pressurizer water level (12%) was not added. This lower limit was related to the previous Safety Injection actuation logic which required a coincident low pressurizer level and low pressurizer pressure trip. This logic was modified as a result of IE Bulletin 79-06A (Ref. 45) to eliminate the coincident low pressurizer level trip (Ref. 46) such that the setpoint is no longer used in an UFSAR Chapter 15 accident analysis. Therefore, the low pressurizer water level setpoint is not required. This is a Ginna TS Category (v.b.3) change.
- x. TS 3.1.1.5.b - The current exception for not requiring the pressurizer heaters and water level setpoints during the RCS hydro test was not added to the new specifications. These hydro tests are performed with RCS temperatures below MODE 3 conditions (i.e., < 350°F). Since the new specification only requires the pressurizer to be OPERABLE in MODES 1, 2, and 3, this exception is no longer required. This is a Ginna TS Category (v.a) change.

- xi. TS 3.1.1.6 - The requirement for the reactor vessel head vents was not added to the new specifications since these vents do not meet the criteria specified in the NRC Policy Statement. This is due to the fact that the vents are used to exhaust noncondensable gases and steam from the RCS which could inhibit natural circulation following an accident with an extended loss of offsite power. However, these vents are not the primary success path and are only used by operators if both pressurizer PORVs are unavailable. These vents are not used in the safety analyses nor were identified as being risk significant in the Ginna Station Level 2 PRA (Ref. 47). This requirement will be relocated from TS to the TRM. The remaining requirements contained within this specification relate to the pressurizer PORVs and their associated block valves which are addressed in TS 3.1.1.4. These requirements were revised as discussed in Section D, items 6.xiii and 6.xiv below. This is a Ginna TS Category (iii) change.
- xii. TS 3.1.1.3.a and 3.1.1.3.b - These requirements were not added to the new specifications since the pressurizer safety valves do not provide overpressurization protection during Cold Shutdown and Refueling conditions. This is provided by the low temperature overpressure protection (LTOP) requirement as specified in current TS 3.15 and new LCO 3.4.12. Since the pressurizer safety valves do not perform a safety function during these low MODES of operation, these requirements were not retained. These changes also supersede those proposed in Reference 60. This is a Ginna TS Category (v.b.4) change.
- xiii. TS 3.1.1.4.a.i and 3.1.1.6 - These were revised to provide separate Required Actions for the PORVs based on the reason for their inoperability. A PORV which is inoperable for automatic functions but capable of manual actuation must be isolated by its block valve consistent with the current requirement. However, a PORV which is incapable of manual cycling is required to be isolated by its block valve within 1 hour and repaired within 72 hours or the plant must initiate a controlled shutdown. In addition, with both PORVs inoperable, a controlled shutdown to MODE 3 conditions with RCS < 500°F must be accomplished within 8 hours. This limit on operation with an inoperable PORV is provided since a SGTR event cannot be mitigated under this condition. The 72 hours for one inoperable PORV is allowed since the second PORV is available. These changes also supersede those proposed in Reference 60. This is a conservative revision and a Ginna TS Category (iv.a) change.

- xiv. TS 3.1.1.4.a.ii and 3.1.1.6 - This was revised to require that one inoperable block valve must be restored to OPERABLE status within 72 hours, both block valves within 7 days, or the plant must initiate a controlled shutdown. This limit on operation with an inoperable block valve is provided since a stuck open PORV cannot be isolated in this condition. The time limits provide adequate time to perform most repairs at power since the valves are located inside containment in the pressurizer cubicle. These changes also supersede those proposed in Reference 60. This is a conservative revision with respect to current requirements and a Ginna TS Category (iv.a) change.
- xv. TS 3.1.1.2 - This was not added since this temperature limit is not required for safe operation. All necessary heatup and cooldown rates are relocated to the PTLR while new LCO 3.4.1 provides limits on RCS pressure, temperature, and flow. This is a Ginna TS Category (v.b.5) change.
- xvi. TS 3.1.1.3.d - A Note was added which allows the pressurizer safety valves to be removed from service above 350°F for the purpose of setting the valves under hot (i.e., ambient) conditions consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.
- xvii. TS 3.1.1.3.c - This was revised to change the pressurizer safety valve lift settings from 2485 psig $\pm 1\%$ to 2485 psig $\pm 2.4\%$, -3% . The valve lift settings are required to be set to within $\pm 1\%$ following testing; however the OPERABILITY tolerances have been revised. The increased OPERABILITY tolerances have been evaluated in the most limiting pressure transients for Ginna Station (i.e., loss of external load and locked rotor events) and found to result in acceptable results with respect to the safety limit values. This change is a result of an event in which the pressurizer safety valves were found to have drifted outside the existing $\pm 1\%$ tolerance band following testing (Ref. 58). The proposed change is within the ASME tolerances of $\pm 1\%$ following testing and $\pm 3\%$ for OPERABILITY. This is a Ginna TS Category (v.b.45) change.

7. Technical Specification 3.1.2

- i. TS 3.1.2.1.a, Figure 3.1-1, and Figure 3.1-2 - The RCS temperature and pressure curves and the RCS heatup and cooldown curves and limits were relocated from technical specifications to the PTLR which is addressed under Administrative Controls. This is a Ginna TS Category (iii) change.

- ii. TS 3.1.2.1.b - The requirement for periodically recalculating the RCS temperature and pressure curves and the RCS heatup and cooldown curves and limits was deleted from technical specifications. A periodic review is already required by 10 CFR 50, Appendix H which does not need to be restated within the technical specifications. This is a Ginna TS Category (ii) change.
- iii. TS 3.1.2.1.c.1 - The time allowed to perform an engineering analysis to determine that the RCS is acceptable to continue operation after a pressure and/or temperature limit is exceeded was increased from 6 hours to 72 hours. A duration of 6 hours is not sufficient time to accomplish the required engineering analysis, especially if the event were to occur during evening or early morning hours with limited staff support immediately available. Since NRC accepted guidance for performing the necessary calculations exists, allowing 72 hours to complete the analyses is appropriate, especially since the duration of event is very limited (i.e., controlled by LCO 3.4.3). This is a Ginna TS Category (v.b.6) change.
- iv. TS 3.1.2.2 - This was not added since this temperature limit is not required for safe operation. All necessary heatup and cooldown rates are relocated to the PTLR while new LCO 3.4.1 provides limits on RCS pressure, temperature, and flow. This is a Ginna TS Category (v.b.5) change.
- v. TS 3.1.2.3 - This was revised to relocate the pressurizer heatup and cooldown rates to the PTLR. The maximum temperature difference between the pressurizer and spray fluid was not added since this limit is controlled by the cooldown curves. These are Ginna TS Category (iii) and (v.c) changes respectively.

8. Technical Specification 3.1.3

- i. TS 3.1.3.1 - This was revised to raise the minimum temperature for criticality from 500°F to 540°F. This change was made to correct a discrepancy between the definition of reactor operating modes and this requirement. Currently, Ginna Station TS 1.2 defines Hot Shutdown as Reactivity $\leq -1 \Delta k/k\%$ and $T_{avg} \geq 540^\circ F$. In order to achieve criticality at 500°F, the Hot Shutdown condition would have to be directly bypassed. A value of 540°F was selected for the new minimum temperature for criticality based on previous operating experience during startup conditions. This is a Ginna TS Category (v.a) change.

- ii. TS 3.1.3.2 - This was not added since LCO 3.4.2 specifies the minimum temperature for criticality. The minimum temperature with respect to the reactor vessel is contained in the PTLR and is below the limit specified in LCO 3.4.2. This is a Ginna TS Category (v.c) change.
- iii. TS 3.1.3.3 - The existing action statement was revised to require that the plant be in MODE 2 with $k_{\text{off}} < 1.0$ within 30 minutes if T_{avg} for one or both RCS loops was $< 540^{\circ}\text{F}$ versus subcritical by an amount equal to or greater than the potential reactivity due to depressurization. The new requirement provides clear and precise instructions to operations and ensures that the plant is quickly brought to a condition in which the LCO is no longer applicable. This is a Ginna TS Category (v.c) change.
- iv. TS 3.1.3.1 - The MTC requirements are moved from the RCS chapter in the Ginna Station TS to the Reactivity Control Systems Chapter. This is a Ginna TS Category (i) change.
- v. TS 3.1.3.1 - This was revised to reference cycle specific MTC requirements in the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The MTC maximum upper limit described in TS 3.1.3.1 remains the same in ITS LCO 3.1.4. This is a Ginna TS Category (iii) change.

9. Technical Specification 3.1.4

- i. TS 3.1.4.4 - This specification was revised to only require shutdown to MODE 3 with $T_{\text{avg}} < 500^{\circ}\text{F}$ within 8 hours versus Cold Shutdown within 40 hours consistent with the LCO Applicability. This is a Ginna TS Category (v.c) change.
- ii. TS 3.1.4.1.c - The limit on secondary coolant activity is now required to be met in MODES 1, 2, 3, and 4 and not just when the reactor is critical or RCS temperature is $> 500^{\circ}\text{F}$. The secondary coolant activity limit is based on a steam line break and the resulting dose consequences. A RCS temperature of $> 500^{\circ}\text{F}$ is based on preventing the MSSVs from lifting following a SGTR (i.e., a RCS temperature of $> 500^{\circ}\text{F}$ is only applicable to primary system activity limits not secondary limits). In addition, if the secondary coolant activity limits are not met, TS 3.1.4.4 requires entering cold shutdown (i.e., MODE 5) within 40 hours. Requiring the secondary coolant activity limits to be met for all of MODE 4 (i.e., RCS is $> 200^{\circ}\text{F}$) provides consistency with NUREG-1431 and the current Required Actions if the limit is exceeded. This is a Ginna TS Category (iv.a) change.

- iii. The time to perform a shutdown if secondary activity is not within limits was changed from 8 hours to 6 hours to reach hot shutdown and 32 hours to 30 hours to reach cold shutdown. These completion times are conservative and provide consistency with the rest of the TS. This is a Ginna TS category (v.a) change.

10. Technical Specification 3.1.5

- i. TS 3.1.5.1.1 - Added a new requirement for the containment sump "A" level or pump actuation per LCO 3.4.15. This leakage detection system replaces the containment humidity detectors and the air cooler condensate flow monitor. The containment humidity detectors do not meet the required leakage rate detection capability of 1.0 gpm within 4 hours as required by Generic Letter 84-04 (Ref. 19). In addition, the containment humidity detectors are recommended by RG 1.45 (Ref. 17) to only be used as an alarm or indirect indication of leakage to containment and not as a separate method of detecting leakage. The remaining leakage detection systems provide adequate monitoring as discussed in the new bases and Section C, item 46. These are Ginna TS Category (v.a) changes.
- ii. TS 3.1.5.1.1 and 3.1.5.1.2 - The RCS leakage detection systems are required to be OPERABLE and RCS LEAKAGE within limits above MODE 4 (200°F) and not 350°F per LCO 3.4.15 and 3.4.13. The increased LCO Applicability will address all MODES in which the RCS is at an increased temperature and pressure. This is a Ginna TS Category (iv.a) change.
- iii. TS 3.1.5.1' - Added a note which allows a change in MODE if either the containment sump monitor or both the containment atmospheric radioactivity monitors are inoperable per LCO 3.4.15. This note is appropriate considering the other instrumentation that is available to monitor RCS leakage. This is a Ginna TS Category (v.b.7) change.

11. Technical Specification 3.1.6

- i. TS 3.1.6 - This entire section was not added since RCS Chemistry does not meet the NRC Policy Statement. RCS Chemistry is controlled by plant procedures and is not required to be addressed within the technical specifications. This requirement is being relocated to the TRM. This is a Ginna TS Category (iii) change.

12. Technical Specification 3.2

- i. TS 3.2.5 - The requirement was revised to require placing a charging pump in pull-stop within 1 hour regardless of the status of the RHR pumps or the MODE. This is a conservative change which provides direct operator guidance to perform an action within a defined time period. Also, these requirements were relocated to the LTOP specification to consolidate all related requirements. The verification of the charging pump status every 12 hours was also not added since the plant is required to be in a depressurized and vented condition within 8 hours which removes the need to isolate a charging pump (i.e., a 1.1 square inch vent can mitigate a charging/letdown mismatch event). These are Ginna TS Category (v.a), (i), and (v.c) changes, respectively.

- ii. TS 3.2.1 and TS 3.2.1.1 - The requirements for the boric acid injection flow paths during cold shutdown and refueling which specifies the number of flow paths that must be OPERABLE were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

- iii. TS 3.2.2 and TS 3.2.4 - The requirements for the boric acid injection flow paths above cold shutdown which specifies the number of flow paths that must be OPERABLE, were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

- iv. TS 3.2.3 and Table 3.2-1 - The requirements for the Boric Acid Storage Tank(s) which specifies the boron concentrations, minimum volume and solution temperature, were not added. The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. Further, the boration system is a non-significant risk contributor to core damage frequency and offsite releases. Therefore, the requirements specified for this system do not satisfy the NRC Final Policy Statement Technical Specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

13. Technical Specification 3.3

- i. TS 3.3.1.1.b and 3.3.1.3 - LCO 3.5.1 Condition A was added which allows 72 hours to restore accumulator boron concentration to within acceptable limits. The ITS bases state that allowing a longer period of time to correct boron concentration is acceptable since the volume of water in the accumulators is the critical feature. Attempting to correct boron concentration within the current 1 hour limit would create a significant burden on the operations staff. Therefore, the current 1 hour LCO was only maintained for accumulator pressure and volume. In addition, the accumulator boron concentration limit was increased to 2100 ppm to support the value used in the accident analysis for the 18 month refueling cycles. An upper limit of 2600 ppm was also added to address chemical considerations of the sump fluid following an accident. This value is also consistent with that used for 18 month refueling cycles. These are Ginna TS Category (v.b.9) and (v.a) changes, respectively.
- ii. TS 3.3.1.1.a and 3.3.1.2 - LCO 3.5.4.A was added which allows 8 hours to restore the RWST boron concentration to within acceptable limits. The ITS bases state that allowing a longer period of time to correct boron concentration is acceptable since it requires a longer period of time to perform this type of adjustment due to the large volume of water contained within the RWST. In addition, the RWST boron concentration limit was increased to 2300 ppm to support the value used in the accident analysis for the 18 month refueling cycles. An upper limit of ≤ 2600 ppm was also added to address chemical considerations of the sump fluid following an accident. This value is also consistent with that used for 18 month refueling cycles. These are Ginna TS Category (v.b.10) and (v.a) changes, respectively.

- iii. TS 3.3.1.1.c - Two notes associated with LCO 3.5.2 were added. The first note allows both SI pump flow paths to be isolated for up to 2 hours to perform pressure isolation valve testing. The ITS bases state that this is acceptable since the isolation valves can be opened from the control room. The second note allows up to 4 hours, or until the RCS cold legs exceed 375°F, to place into service ECCS pumps declared inoperable due to LTOP considerations. This note was added since the LTOP setpoint of 330°F is very close to the Mode 3 definition of $\geq 350^\circ\text{F}$. As described in the ITS bases, this note provides operator flexibility to restore the inoperable pump to OPERABLE status. These are Ginna TS Category (v.b.11) changes.
- iv. TS 3.3.1.5.d - This was revised and used as a note for LCO 3.5.2. The specification now only allows 878B and 878D to have power installed during MODE 3 for the specific purpose of performing pressure isolation valve testing. Isolation valves 878A, 878C, 896A, 896B and 856 must now have DC power removed above MODE 3 or both trains of ECCS will be declared inoperable. This change was made since there is no regularly scheduled testing of 878A, 878C, 896A, 896B, and 856 above 350°F. This is a Ginna TS Category (v.a) change.
- v. LCO 3.5.3 was added which requires one train of SI and RHR during MODE 4. This new requirement is being added to address low probability accidents which may occur during this mode of operation. This is a Ginna TS Category (iv.a) change.
- vi. TS 3.3.1.1.b - The current exception for not requiring the accumulators during hydro tests was not added to the new technical specifications. These hydro tests are performed with RCS temperatures below MODE 3 conditions (i.e., $< 350^\circ\text{F}$). Since the new specification only requires the accumulators when RCS pressure is > 1600 psig during MODE 3, this exception is no longer required. This is a Ginna TS Category (vi) change.
- vii. TS 3.3.1.1.b - The bases for TS 3.3 were revised to update the specified water volume contained in the accumulator with respect to the 50% and 82% levels. The required levels specified in TS 3.3.1.1.b have not been changed, only the corresponding water volumes provided in the bases. The new values are consistent with those used in the accident analysis (see COLR, Table 1). This is a Ginna TS Category (v.c) change.

- viii. TS 3.3.1.1.g - Motor operated isolation valves 851A and 851B were added to new SR 3.5.2.1 since these valves must remain open with AC power removed to ensure the availability of Containment Sump B to the RHR system following a LOCA. The addition of these valves is a conservative change. This is a Ginna TS Category (v.a) change.
- ix. TS 3.3.1.1.h - Check valves 877A, 877B, 878F, 878H, and motor operated isolation valves 878A and 878C were added to this requirement since the valves are required to be tested as PIVs by current Ginna Station TS 4.3.3.3. This provides a more complete specification and is a Ginna TS Category (v.a) change. The listing of valves was also relocated to the bases. This is a Ginna TS Category (iii) change.
- x. TS 3.3.1.1.h and 3.3.1.5 - These requirements were revised to require PIVs to be OPERABLE in MODES 1, 2, 3, and 4 and not just above 350°F (i.e., in MODE 3 and above). Therefore, the plant must now enter MODE 5 within 36 hours if the Required Actions cannot be accomplished. This is a conservative revision which expands the LCO Applicability. This is a Ginna TS Category (iv.a) change.
- xi. TS 3.3.1.5.e - The current requirement allows 12 hours to repair a leaking check valve if the in-series motor operated isolation valve is closed. This was revised to specify that a leaking PIV (check valve or motor operated) must be isolated within 4 hours with a leak tested valve, and that a second leak tested valve must be closed within 72 hours. This is generally a conservative change since a time limit is now specified for isolating the leaking valve and the second isolation valve must now be leak tested. The only exception is that 72 hours is now provided to perform repairs versus 12 hours. The existing allowed repair time is insufficient to perform most leakage repairs and would most likely require a reactor shutdown. Since there are three isolation valves for several flow paths, and the LCO applicability has been expanded to include MODE 4, this change is considered acceptable. This is a Ginna TS Category (v.a) change.
- xii. TS 3.3.1.7 and 3.3.1.8 - The exception for allowing the SI pumps to be OPERABLE during DG load and safeguard sequence testing was not added since the new bases allow the pumps to be OPERABLE if a discharge isolation valve is locked closed. Therefore, this exception is not required. Also, these requirements were relocated to the LTOP specification to consolidate all related requirements. These are Ginna TS Category (v.c) and (i) changes, respectively.

- xiii. TS 3.3.1.7.1 and 3.3.1.8.1 - These specifications were converted into Surveillance Requirements consistent with the ITS format and relocated to the LTOP specification to consolidate all related requirements. This is a Ginna TS Category (i) change.
- xiv. TS 3.3.1.8.2 - This requirement was not added since the new bases list the criteria for ensuring that a SI pump is incapable of injecting into the RCS. Limiting the operation to one SI pump when the PORVs provide the RCS vent path is not necessary if the isolation device requires two separate actions before providing an injection path to the RCS. Therefore, operating multiple SI pumps will not pose any threat to overpressurizing the RCS with this isolation. This is a Ginna TS Category (v.c) change.
- xv. TS 3.3.2.2 - This was revised to allow both post-accident charcoal filter trains (including the CRFC units which supply them) to be inoperable for up to 72 hours if both containment spray (CS) trains are OPERABLE. This change provides consistency with the accident analyses which demonstrate that either two CS trains, one CS train and one post-accident charcoal filter train, or two post-accident charcoal filter trains are adequate to remove radioactive iodine from the containment atmosphere following a DBA (i.e., each CS train and post-accident charcoal filter train provides 50% of the required iodine removal requirements). However, two CS trains cannot be inoperable since at least one train must operate for containment pressure and temperature control. In addition, two CRFC units can now be removed from service for up to 7 days since the accident analyses only credit two of the four cooling units as being OPERABLE with respect to containment pressure and temperature control. Finally, with one or two CRFC units inoperable and not restored within 7 days, the plant has only 36 hours to reach MODE 5 versus 84 hours due to the importance of maintaining containment pressure and temperature control. These are Ginna TS Category (v.b.12) changes.
- xvi. TS 3.3.3.1 - This was revised to specify that the CCW loop header must also be OPERABLE. The loop header is defined as the section of piping from the discharge of the heat exchangers to the first isolation valve of each supplied component. The loop header then continues from the last isolation valve on the discharge of the supplied component to the suction of the pumps. This is a Ginna TS Category (v.a) change.

- xvii. TS 3.3.3.1 - This was revised to allow one CCW heat exchanger to be removed from service for up to 31 days. As discussed in Section C, item 82.i above, the CCW heat exchangers are 100% redundant and are separated from the CCW pump trains by a section of common piping. Since there is no signal active failure which could fail the redundant heat exchanger, 31 days is considered acceptable. Also, there is only one loop header such that a passive failure of the loop header, or the remaining OPERABLE heat exchanger, has the same consequences. This is a Ginna TS Category (v.b.13) change.
- xvii. TS 3.3.3.2 - This was revised to allow 72 hours (versus 24 hours) to restore an inoperable CCW pump before requiring a plant shutdown. However, the plant is no longer allowed to remain at Hot Shutdown for 48 hours before requiring additional cooldown to Cold Shutdown conditions. As such, the total time in which a CCW pump can remain inoperable remains the same (i.e., 72 hours) but the plant is not required to begin cooldown activities after 24 hours. The only safety related functions supported by the CCW System are with respect to the RHR, SI, and CS Systems, which all allow 72 hours to restore an inoperable train. Therefore, this change provides consistency within the new specifications. This is a Ginna TS Category (v.c) change.
- xviii. TS 3.3.4.1 - This was revised to require that the six sets of motor operated isolation valves used in the SW System to be OPERABLE for the SW System to be considered OPERABLE. Credit is taken for these valves to isolate the nonessential and nonsafety related components within the SW System following a coincident safety injection and undervoltage signal. This is a conservative change which provides a clarification to licensed personnel. This is a Ginna TS Category (v.a) change.

xix. TS 3.3.4.2 - This was revised to allow one SW train comprised of two pumps and six motor operated valves supplied by the same electrical train to be inoperable for 72 hours before requiring a plant shutdown. Since the SW trains are 100% redundant, removing one of two trains only affects redundancy and does not place the plant outside the accident analyses. Since most other safety functions allow 72 hours for one train to be inoperable (e.g., ECCS trains), this change provides consistency within the new specifications. In addition, this specification was revised to address the scenario if all SW pumps or the SW loop header are inoperable. In this condition, immediate action must be initiated to restore one SW pump or the loop header to OPERABLE status; however, it may not be prudent to exit the MODE of Applicability since the SW System is required in MODE 5 for decay heat removal. Instead, Required Actions have been provided to require a cooldown to MODE 5 unless the CCW system is incapable of supporting RHR. In MODE 4, AFW is providing for decay heat removal. If AFW were lost, additional time is required before RHR (and consequently SW) would be required. These are Ginna TS Category (v.b.47) changes.

xx. TS 3.3.5.1 - This was revised to require the control room emergency air treatment system (CREATS) to be OPERABLE in MODES 1 through 6 and during movement of irradiated fuel assemblies instead of only when RCS is $\geq 350^{\circ}\text{F}$. Current Ginna Station TS 3.5.6 requires that the control room HVAC detection system (i.e., chlorine, ammonia, and radioactivity monitors) be OPERABLE at all times. However, the filtration system is only required to be OPERABLE above 350°F . The filtration system is designed to ensure that dose rates to operators are within the guidelines of GDC 19 in the event of an accident. While dose rates to operators is expected to be lower when the RCS is $< 350^{\circ}\text{F}$, no current analyses exist under these conditions. In addition, failures of the waste gas decay tanks can still occur below 350°F which also require control room isolation. Therefore, the MODE of Applicability was revised to provide consistency within the specifications and the accident analyses. This is a Ginna TS Category (iv.a) change.

- xxi. TS 3.3.5.2 - This was revised to provide requirements for an inoperable filtration train and inoperable dampers. The CREATS dampers isolate the control room in the event of a radiological event while the filtration train filters the control room atmosphere following isolation. The new specification continues to allow the filtration train to be inoperable for 48 hours before requiring a shutdown or placing the control room in the emergency radiation mode (i.e., CREATS Mode 6). If one of the two redundant dampers in each outside air flow path is inoperable, the new specifications allow 7 days to restore the damper to OPERABLE status similar to restoring one train of redundant CREFS in NUREG-1431. If both dampers are inoperable, the plant must enter LCO 3.0.3 since the control room can no longer be isolated. If both dampers are lost in MODES 5 or 6, or during fuel movement, then fuel movement and CORE ALTERATIONS must be suspended immediately. These changes provide consistency with the accident analyses and NUREG-1431. These are Ginna TS Category (v.a) changes.

14. Technical Specification 3.4

- i. TS 3.4.1 - This was revised to specifically require that all MSSVs be tested prior to entering MODE 2 versus the current wording which allows the MSSVs to be removed for testing at any time. This change is consistent with current operating practices and ensures that the MSSVs are OPERABLE before the reactor goes critical but allows the MSSVs to be tested under hot conditions (i.e., $\geq 350^{\circ}\text{F}$). In addition, the MSSV setpoints were added to the new specification since these are assumptions within the accident analyses. These are Ginna TS Category (v.a) changes.

- ii. TS 3.4.2.1.b - This was revised to be consistent with the accident analysis assumptions as discussed in the new bases. Essentially, the accident analyses treat the preferred AFW System as four trains (i.e., two motor driven trains and two turbine driven trains) such that each SG receives flow from two AFW trains. Therefore, the failure of both motor driven trains or the turbine driven train (or both flowpaths) has the same consequence (i.e., loss of one train to each SG). Since the turbine driven train is allowed to be inoperable for up to 72 hours per TS 3.4.2.2.a (and NUREG-1431), this specification was revised to allow both motor driven AFW pumps to be inoperable for up to 72 hours. In addition, if both AFW trains to a common SG are inoperable, the new specifications allow 4 hours to restore at least one train before requiring a controlled cooldown. A time limit for being in this configuration is necessary since no AFW would be available in the event of a HELB which affects the only SG able to receive AFW. Requiring an immediate cooldown in this configuration is not considered prudent since AFW provides for decay heat removal in lower MODES. These are Ginna TS Category (v.b.14) and (v.a) changes, respectively.
- iii. TS 3.4.2.3 - This was revised to require that the SAFW cross-tie be available when the SAFW System is required to be OPERABLE. This change is required since the accident analyses credit the use of the cross-tie for HELBs with a failure of one SAFW pump. Each cross-tie motor operated valve is considered part of the SAFW train which shares the same electrical power source. This is a Ginna Station TS Category (v.a) change.
- iv. TS 3.4.3 - The requirement for SW suction for the AFW and SAFW pumps were relocated to the LCO for these pumps. The CSTs provide the preferred source of condensate to the preferred AFW pumps while the SW System is the safety related source for both the preferred and standby AFW systems. The relocation of the need for a SW supply to the AFW pumps within technical specifications does not reduce the requirement. Instead, the change provides consistency within the new specifications and is easier for licensed personnel to understand. This is a Ginna TS Category (i) change.

- v. TS 3.4.3 - This was revised to require that a backup source of condensate be verified within 4 hours when the CSTs are inoperable versus demonstrating the operability of the SW System. Specifying a time limit for verifying the backup condensate source is a conservative change which now provides a clear and concise requirement for plant operators. Revising the Actions to allow any alternate source to be used as a backup source provides additional operational flexibility since other condensate sources than the SW System can be used if necessary. These sources are described in the bases for new LCO 3.7.6. These changes are consistent with NUREG-1431 and are Ginna TS Category (v.a) changes.

15. Technical Specification 3.5

- i. The following changes were made to TS 3.5.1 or Table 3.5-1:
 - a. Table 3.5-1, Columns 1, 2, and 3 - The columns for the "Total Number of Channels," the "Number of Channels to Trip," and the "Minimum Operable Channels" were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the OPERABILITY of the instrumentation and were relocated to the bases or are adequately described in the UFSAR. This is a Ginna TS Category (iii) change.
 - b. Table 3.5-1, Column 6 - The column for the "channel operable above" was revised consistent with the changes to the Mode table definitions in ITS Chapter 1.0. Changes to the Applicability different from those discussed in Chapter 1.0 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (vi) change.
 - c. Table 3.5-1, Functional Unit #15 - The trip Function was not added to the new specifications. Removal of this trip function is justified in Reference 44 which shows that based on the offsite power system configuration, this trip Function is not applicable to Ginna Station. Therefore, this trip Function was relocated to the TRM. This is a Ginna TS Category (iii) change.

- d. Table 3.5-1, Action Statement #1 for Functional Unit #1 - This action was revised to add requirements for operability of the Manual Reactor Trip function in Modes 3, 4, and 5 when the rods are not fully inserted and the rod control system is capable of rod withdrawal (LCO 3.3.1, Condition C). These actions ensures the plant is placed in a condition in which the trip function is no longer required for the associated modes of operation. This is a Ginna TS Category (vi) change.
- e. Table 3.5-1, Functional Unit #11 - This was revised to add the requirements for Turbine Trip on Turbine Stop Valve Closure since this was not in the CTS. The required actions with inoperable channels are the same as that for the Turbine Trip on Low Autostop Oil Pressure. This is a Ginna TS Category (iv.a) change.
- f. Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- g. Table 3.5-1, Action Statement #2 for Functional Units #2 ("low setting" and "high setting"), #5, #6, and #7 - This action was revised to allow an inoperable channel to be bypassed for up to 4 hours (rather than 2 hours) during surveillance testing. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- h. Table 3.5-1, Column 4 - This requirement was revised to associate the permissive (or bypass) details with the specific permissive (or interlock) numbers and to clarify the applicability of the Function with an associated Mode (see ITS Table 3.3.1-1, FU #16). The details of the permissible bypass conditions for the associated Functions are discussed in the UFSAR and ITS Bases. Changes to the Applicability of a Functional Unit different from those discussed in Column 4 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (v.c) change.

- i. Table 3.5-1, Channel Operable above column for Functional Units 7, 10, 14 and 15. This was revised to change the MODE of Applicability to 8.5% RTP versus 5% RTP. The permissive which enables these functions to be OPERABLE is set at 8.5% RTP per CTS 2.3.2.1. Therefore, this change provides consistency within the CTS. This is a Ginna TS Category (v.a) change.
- j. Table 3.5-1, Action Statement #3 and #4 for Functional Units #2, #3, and #4 - These actions were revised to clarify the applicability of the intermediate range neutron flux and source range neutron flux to correspond to the specific permissives. The NIS intermediate range neutron flux channels must be OPERABLE when the power level is above the capability of the source range and below the capability of the power range. The associated Required Actions ensure the plant is no longer in the applicable condition through controlled power adjustments and taking into account the low probability of an event during the period that may require the protection of the NIS trip. This change supersedes that proposed in Reference 61. This is a Ginna TS Category (v.a) change.
- k. Table 3.5-1, Action Statement #4 for Functional Unit #4 - This action was revised to clarify the Applicability and add associated Required Actions for inoperable SRMs. For Mode 2 below the permissive and only one SRM OPERABLE, positive reactivity additions must stop immediately and the inoperable channel restored in 48 hours consistent with current TS. However, with two SRMs inoperable the plant would be required to immediately open the RTBs. For Modes 3, 4, and 5, with the CRD incapable of rod withdrawal or all rods not fully inserted, an additional action was added that requires the performance of a SDM verification. These clarifications and additional restriction ensure the plant is no longer in the applicable condition or is in a more stable condition. This is a Ginna TS Category (iv.a) change.

- l. Table 3.5-1, Action Statement #5 for Functional Units #8, #9, #10 ("low flow in one loop"), #11 and #13 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- m. Table 3.5-1, Action Statement #5 for Functional Units #8, #9, #10 ("low flow in one loop"), #11 and #13 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- n. Table 3.5-1, Action Statement #6 for Functional Units #10 ("low flow in both loops"), #14 and #15 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- o. Table 3.5-1, Action Statement #6 for Functional Units #10 ("low flow in both loops"), and #14 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

- p. Table 3.5-1, Functional Unit #16 - This was revised to relocate the QPTR Monitor OPERABILITY requirements to Chapter 3.2. In addition, requirements were added to verify with a calculation that the QPTR is within limits every 24 hours when the Quadrant Power Tilt Monitor is inoperable and THERMAL POWER is $< 75\%$ RTP and to verify with a full core flux map that the core power distribution is acceptable every 24 hours when the Quadrant Power Tilt Monitor is inoperable and THERMAL POWER is $\geq 75\%$ RTP. These are Ginna TS Category (i) and (iv.a) changes, respectively.
- q. Table 3.5-1, Functional Unit #17 - The trip function requirement for the Circulation Water Flood Protection was not added. The Circulation Water Flood Protection instruments only provide an anticipatory turbine trip and is not assumed in the Ginna Station safety analysis. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- r. Table 3.5-1, Functional Units #18 and #19 - The Functional Unit applicability was revised to require the instruments to be applicable in all modes associated with DG operability. This ensures that the DG can perform its function on a loss of voltage or degraded voltage to the 480 V buses. This is a Ginna TS Category (iv.a) change.
- s. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This Completion Time is sufficient to allow restoration of the channel and takes into account the redundancy of the trip channels, and the low probability of an event requiring a LOP start occurring during this interval. This is a Ginna TS Category (v.b.16) change.

- t. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel (consistent with LCO 3.0.5) in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with the associated logic. Bypassing the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. Additionally, a note was added clarifying that entry into the associated Conditions and Required Actions can be delayed for up to 4 hours for performance of required surveillance. Entering DG actions during testing is not necessary since the Completion Times for an inoperable DG is much greater than the time to perform the SR (72 hours vs 6 hours). The SR Note time of 4 hours takes into account the redundancy of the trip channels and the low probability of an event requiring a LOP start occurring during this interval. This is a Ginna TS Category (v.b.17) change.
- u. Table 3.5-1, Action Statement #7 for Functional Units #18 and #19 - This action was revised to replace the current shutdown actions with a requirement to restore channels to an OPERABLE status or to enter the applicable conditions for an inoperable DG. The actions of new LCO 3.8.1 and LCO 3.8.2 provide for adequate compensatory actions to assure plant safety. The loss of the minimum required loss of voltage or degraded voltage channels (one bus) should result in actions that are no more restrictive than actions for the loss of one DG. This is a Ginna TS Category (iv.b.1) change.
- v. Table 3.5-1, Functional Unit #18 and #19 - The number of channels was reformatted to require only two undervoltage channels per bus versus two channels of the loss of voltage function and two degraded voltage function per bus. The bus undervoltage design is a one-out-of-two taken twice logic such that one degraded voltage channel and one loss of voltage channel comprise each of the two undervoltage channels. However, due to the system design, if either of the degraded voltage or loss of voltage functions is inoperable, the entire undervoltage channel must be tripped (i.e., both the degraded voltage and loss of voltage functions are tripped). This change provides greater clarity to the operators without any reduction in the system requirements. This is a Ginna TS Category (v.b.18) change.

- w. LCO 3.3.1, Table 3.3.1-1, Function #10 was added for the RCP Breaker Position. This function anticipates the Reactor Coolant Flow - Low trips by monitoring each RCP breaker position to avoid RCS heatup that would occur before the low flow trip actuates. The function ensures that protection is provided against violating the DNBR limit due to loss of flow in either a single loop or two loop configuration. This is a Ginna TS Category (iv.a) change.
- x. LCO 3.3.1, Table 3.3.1-1, Function #14 was added for the SI Input from ESFAS. This function ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. A reactor trip is initiated every time an SI signal is present. This is a Ginna TS Category (v.a) change.
- y. Table 3.5-1, Functional Unit #20 and associated Action Statement #14 - This requirement was reformatted to separately denote the Reactor Trip Breakers, the Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, and the Automatic Trip Logic functions (LCO 3.3.1, Table 3.3.1-1, Functions #15, #16, and #17). This is a Ginna TS Category (vi) change.
- z. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Automatic Trip Logic) - This action was revised to allow 6 hours to restore the channel to OPERABLE status in Modes 1 and 2 prior to initiating a plant shut down to Mode 3 (new LCO 3.3.1, Condition R). The restoration time of 6 hours is reasonable considering that the remaining OPERABLE channel is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.18) change.
- aa. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker) - This action was revised to allow 1 hour to restore the RTB to OPERABLE status in Modes 1 and 2 prior to initiating a plant shut down to Mode 3 (new LCO 3.3.1, Condition T). The restoration time of 1 hour is reasonable considering that the remaining OPERABLE RTB is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.19) change.

- bb. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Automatic Trip Logic) - This action was revised to allow 48 hours to restore the channel to OPERABLE status in Modes 3, 4, and 5 prior to initiating action to open the RTBs (new LCO 3.3.1, Condition W). The restoration time of 48 hours is reasonable considering that the remaining OPERABLE channel is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.20) change.
- cc. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker) - This action was revised to allow 48 hours to restore the breaker to OPERABLE status in Modes 3, 4, and 5 prior to initiating action to open the RTBs (new LCO 3.3.1, Condition W). The restoration time of 48 hours is reasonable considering that the remaining OPERABLE breaker is adequate to perform the safety function and given the low probability of an event during this interval. This is a Ginna TS Category (v.b.20) change.
- dd. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms) - This action was revised to only allow 1 hour to open the RTBs following the action to restore the RTB to OPERABLE status in Modes 3, 4, and 5 (new LCO 3.3.1, Condition X). The current Ginna Station TS allows 6 hours to perform this action but takes into account a shut down from Modes 1 and 2. The 1 hour provides sufficient amount of time to accomplish the action in Modes 3, 4, and 5 in an orderly manner. This is a Ginna TS Category (v.a) change.
- ee. Table 3.5-1, Action Statement #14 for Functional Unit #20 (Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms) - This action was revised to specify a limit of 2 hours to bypass the RTB for surveillance testing and 6 hours to bypass the RTB for maintenance on undervoltage or shunt trip mechanisms (new LCO 3.3.1, Condition T, Notes 1 and 2). The current Ginna Station TS for bypassing during maintenance does not specify a time limit. The ITS would set a limit on this time. This is a Ginna TS Category (iv.a) change.

- ii. The following changes were made to TS 3.5.2, Table 3.5-2, or Table 3.5-4:
- a. TS 3.5.2.2, 3.5.2.3 and Table 3.5-2, Columns 1, 2, and 3 - The details describing the operability acceptance criteria for Trip Setpoints including the columns for the "Total Number of Channels," the "Number of Channels to Trip," and the "Minimum Operable Channels" were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the operability of the instrumentation and were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.
 - b. Table 3.5-2, Column 6 - The column for the "Channel Operable Above" was revised consistent with the changes to the Mode table definitions in ITS Chapter 1.0. Changes to the Applicability different from those discussed in Chapter 1.0 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (vi) change.
 - c. Table 3.5-2, Functional Unit #1.b - The Mode of Applicability was revised to be RCS > 200°F. The SI High Containment Pressure Function is used to actuate containment isolation below 350°F such that this Function must be operable. The Manual SI Function does not actuate Containment Isolation while the remaining functions are blocked when RCS pressure is < 2000 psig. This is a Ginna TS Category (v.a) change.
 - d. Table 3.5-2, Functional Units #1.c and #1.d - The notes or remarks which describe operational details for the Pressurizer Pressure interlock, were reformatted as Mode Applicabilities and default conditions in the new specifications. A new SR 3.3.2.6, was added to specifically denote the operability requirements for the Pressurizer Pressure interlock. This is a Ginna TS Category (iii) change.

- e. Table 3.5-2, Action Statement #9 for Functional Units #1.b, #1.c, #1.d, #3.b.i, #5.c and #6.b - This action was revised to replace the current limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- f. Table 3.5-2, Action Statement #9 for Functional Units #1.b, #1.c, #1.d, #3.b.i, #5.c and #6.b - This action was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- g. LCO 3.3.2, Functional Units #1.b, #2.b, #3.b, #4.b, #5.a, and #6.a, "Automatic Actuation Logic and Actuation Relays," were added for the ESFAS Instrumentation. Actuation logic consists of all circuitry housed within the actuation subsystems, including relay contacts responsible for actuating the ESF equipment. This is merely a presentation change to the Technical Specifications as this logic circuitry is assumed within the operability of the specific Functions. Additionally, the automatic actuation logic and actuation relays for various Functions are required OPERABLE in Mode 4 to support system level manual initiation. This is a Ginna TS Category (iv.a) change.
- h. Table 3.5-2, Action Statement #12 for Functional Unit #3.c - The action associated with this Function was revised to allow an inoperable channel to be placed in the tripped condition within 48 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

- i. Table 3.5-2, Action Statement #11 for Functional Unit #2.b - The action associated with this Function was revised to replace the limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- j. Table 3.5-2, Action Statement #11 for Functional Unit #2.b - The action associated with this Function was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 2 hours). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- k. Table 3.5-2, Functional Unit #3.a and #3.f - The actions for an inoperable Manual Initiation channel for the AFW and SAFW Systems was revised from restoring operability in 48 hours to declaring the associated pump inoperable. The Manual Initiation channels for these functions actually consist of switches in the control which only actuate one pump train. There is no switch for complete actuation of all AFW or SAFW pumps. Therefore, entering the pump inoperability requirements is consistent with the actions if the AFW pump were declared inoperable. This is a Ginna TS Category (v.b.51) change.
- l. Table 3.5-2, Action Statement #12 for Functional Units #3.b.ii, #3.c, #5.a, and 5.b - The action associated with these Functions was revised to replace the limitation of operation (tied to the next channel functional test of an OPERABLE channel) to allow the bypassing of an inoperable channel for up to 4 hours in order to perform surveillance testing of other channels. The current requirement limits the ability to perform channel functional tests on OPERABLE channels for Functional Units with two-out-of-three logic. Providing a note to bypass the inoperable channel provides a sufficient timeframe to perform the required surveillance testing in a safe and orderly manner. This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.

- m. Table 3.5-2, Action Statement #12 for Functional Units #3.b.ii, #5.a, and 5.b - The action associated with these Functions was revised to allow an inoperable channel to be placed in the tripped condition within 6 hours (rather than 1 hour). This change is discussed and justified in Reference 30. This is a Ginna TS Category (v.b.15) change.
- n. Table 3.5-2, Action Statement #6 for Functional Unit #3.e - The action associated with this Function was revised to a more restrictive restoration time of 48 hours for an inoperable channel rather than placing the channel in the tripped condition within one hour. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 30. This is a Ginna TS Category (iv.a) change.
- o. Not used.
- p. Table 3.5-2, Functional Unit #4.2 - The requirements for the Containment Ventilation Isolation (CVI) Manual Initiation Function were not added. The current TS are misleading in that there is no manual CVI initiation function. Instead, CVI is manually initiated by the Manual CS function. The removal of this requirement provides consistency within the TS and greater clarify to the operators. This is a Ginna TS Category (v.c) change.
- q. Table 3.5-4, Functional Units #1.b, #1.d, and #2.b - These Functional Unit Allowable Values were revised to reflect the actual values used in the accident analyses. This is a Ginna TS Category (v.c) change.
- r. Table 3.5-4, Functional Units #7.a and #7.b - The Trip Setpoint for the loss of voltage and degraded voltage functions were revised to provide a minimum value. Criteria for the establishment of equivalent values based on measured voltage versus relay operating time was relocated to the bases for new LCO 3.3.4). This is a Ginna TS Category (iii) change.
- s. Table 3.5-4, Notes 1 and 2 for Functional Units #6.a and #6.c - The notes which describe design details for the Steam Generator Water Level - Low Low Function and Loss of 4 kV Function were not added. These details are relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.

t. Table 3.5-2, Functional Unit #4.2b and Action 8 to Table 3.5-5 - The actions for an inoperable CVI radiation monitor were revised to allow 4 hours to isolate the affected penetration. Current TS Table 3.5-2 does not provide a time limit, only that the valves are to be closed while Table 3.5-5 provides 1 hour to perform this action. The time limit is consistent with NUREG-1431 and the Completion Times for an inoperable containment isolation valve which the CVI signal actuates. This is a Ginna TS Category (v.b) change.

iii. The following changes were made to TS 3.5.3 or Table 3.5-3:

- a. TS 3.5.3.2, TS 3.5.3.3, and Table 3.5-3, Columns 1 and 2 - The columns for the "Total Required Number of Channels," and the "Minimum Channels Operable," were not added for each of the functional units. The columns were replaced with a new column denoting "Required Channels." System design and operational details are not directly related to the operability of the instrumentation and were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.
- b. TS 3.5.3.2 - The restoration time requirement of 7 days for one inoperable channel (for Functions with two channels) was revised to 30 days. The 30 day Completion Time was revised based on industry operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument, and the low probability of an event requiring PAM instrumentation during this interval. This is a Ginna TS Category (v.b.21) change.
- c. TS 3.5.3.2 - The action for one channel inoperable for more than 7 days (for Functions with two channels) was revised from requiring a plant shutdown to requiring a Special Report. Due to the passive function of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out-of-service instrumentation. This is a Ginna TS Category (v.b.21) change.

- d. TS 3.5.3.3 - The restoration time requirement of 48 hours for two inoperable channels was revised to 7 days. The 7 day Completion Time was revised based on industry operating experience and takes into account the availability of alternate means to obtain the required information and the low probability of an event requiring PAM instrumentation during this interval. This is a Ginna TS Category (v.b.21) change.
- e. Table 3.5-3 - The Post Accident Monitoring Instrumentation Functions required by this specification were revised to include only RG 1.97, Type A and Category I variables. These functions are denoted in UFSAR Table 7.5-1 and have been previously reviewed and approved by the NRC (Ref. 59). This is a Ginna TS Category (iv.a) change.
- iv. TS 3.5.4 and Table 3.5-6 - The requirements for radiation accident monitoring instrumentation, provided to monitor radiation levels in selected plant locations following an accident, were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM. This is a Ginna TS Category (iii) change.
- v. TS 3.5.6.1 - The requirements for the chlorine gas and ammonia gas instrumentation monitors for control room habitability were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.
- vi. LCO 3.3.5, Conditions B and C, were added for the Control Room Emergency Air Treatment System (CREATS) actuation instrumentation. These new requirements specify Required Actions for various modes of operation when the CREATS isolation dampers cannot be placed in the emergency radiation protection mode. This is a Ginna TS Category (iv.a) change.

- vii. TS 3.5.6.2 - The requirement for one detection system inoperable has been revised to allow more than one channel inoperable with an action to isolate the control room in one hour. Even with a loss of Function of the automatic actuation logic, the CREATS may still be capable of being manually isolated within 1 hour and performing its safety function. This is a Ginna TS Category (v.c) change.
- viii. TS 3.5.5 and Table 3.5-5 - The requirements for radioactive effluent monitoring instrumentation which ensures that the limits of TS 3.9.1.1 and 3.9.2.1 are not exceeded were not added (except for R-11 and R-12 which support Containment Ventilation Isolation). No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively.
- ix. Table 3.5-5 - The Mode of Applicability for R-11 and R-12 operability was revised from "during shutdown purges" to MODES 1, 2, 3, and 4, and during MODE 6 when required by LCO 3.9.3, "Containment Penetrations." This is a conservative change which ensures that the gaseous and particulate radiation monitors are operable in all Modes in which the mini-purge or shutdown purge systems can be used. This is a Ginna TS Category (v.a) change.

16. Technical Specification 3.6

- i. TS.3.6.1 - The text allowing closed containment isolation valves to be opened on an intermittent basis under administrative controls was relocated to a LCO Note consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.
- ii. TS 3.6.2 - The Applicability for maintaining containment pressure within limits was revised from reactor criticality to MODE 4. This change is necessary to provide consistency with the requirements for containment integrity (i.e., LCO 3.6.1) since exceeding these pressure limits could result in a overpressure of containment if an accident were to occur. This is a Ginna TS Category (iv.a) change. Also, the time allowed to restore containment pressure was changed from 24 hours to 8 hours. This change was made due to the short period of time it normally takes to restore containment pressure to within limits. This is a Ginna TS Category (v.a) change.

- iii. TS 3.6.3 - Three new requirements were added. The first requires that a penetration with both containment boundaries inoperable be isolated within 1 hour versus 4 hours. This change provides consistency with TS 3.6.1 since containment integrity is potentially violated. As such, verification of continued acceptable containment leakage must be initiated immediately if both barriers are declared inoperable. In addition, new requirements with respect to an inoperable airlock (including the use of an airlock with an inoperable door or interlock mechanism) and containment mini-purge penetrations with isolation valves that exceed their leakage rate acceptance criteria were added. The new requirement for the airlocks specifies that an inoperable airlock door (including an inoperable interlock mechanism) must be isolated within 1 hour and locked closed within 24 hours. However, a dedicated individual can be used in the case of an inoperable interlock mechanism to allow entry and exit through the airlock. The new specification provides specific Required Actions in the event that current Ginna Station TS 4.4.2.3.c is exceeded. The new requirement for the mini-purge penetrations specifies that the affected penetration must be isolated within 24 hours if an isolation valve exceeds its leakage limit. These new requirements provide added assurance that penetrations which can provide direct access to the outside environment are addressed quickly when their isolation barriers become inoperable. This is a Ginna TS Category (iv.a) change.
- iv. TS 3.6.3 - The use of a closed system to isolate an inoperable containment isolation barrier for up to 72 hours was added to this specification. Consequently, a closed system which must be OPERABLE to meet this specification can be used to isolate a failed isolation barrier for a limited period of time. Also, isolation devices located outside containment that were used to isolate a failed containment isolation valve are required to be verified closed once every 31 days. For isolation devices inside containment, they must be verified closed upon entry into MODE 4 from MODE 5 if it has not been performed within the last 92 days. These are Ginna TS Category (v.b.22) changes.
- v. TS 3.6.5 - The reasons for opening the mini-purge valves above 200°F were relocated to the bases for ITS 3.6.3 since these do not meet any of the four criteria and do not specify any Required Actions. Operation of the Mini-Purge System is performed under procedures such that its use is strictly controlled. Placing this information in the bases also provides similar control under 10 CFR 50.59 (i.e., the Bases Control Program). This is a Ginna TS Category (iii) change.

- vi. TS 3.6 - A new requirement was added which specifies that the average containment air temperature shall be $\leq 120^{\circ}\text{F}$ above MODE 5. This temperature limit is necessary to ensure that the resulting containment temperature following a DBA is within the assumptions used for environmental qualification of components within containment. If the average containment air temperature is $> 120^{\circ}\text{F}$, it must be restored within 24 hours. This is a Ginna TS Category (iv.a) change.
- vii. TS 3.6 - A new requirement was added which requires the hydrogen recombiners to be OPERABLE in MODES 1 and 2. The hydrogen recombiners are assumed in the accident analyses to be used to prevent a hydrogen explosion within containment that could overpressurize the containment structure. The new LCO allows 30 days to restore an inoperable recombiner and 7 days to restore two inoperable recombiners if the Mini-Purge System is OPERABLE. In addition, the plant can enter MODES 1 and 2 with an inoperable hydrogen recombiner. This is a Ginna TS Category (iv.a) change.
- viii. TS 3.6.4.1 and TS 3.6.4.3 - The Applicability for the hydrogen monitors was revised to include Mode 3 requirements. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in Modes 1, 2, and 3. This is a Ginna TS Category (iv.a) change.
- ix. TS 3.6.4.2 - The action for one channel inoperable for more than 30 days was revised from requiring a plant shutdown to requiring a Special Report. Due to the passive function of these instruments and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out-of-service instrumentation. This is a Ginna TS Category (v.b.21) change.
- x. TS 3.6.1.b - The requirement describing the specific applicability for containment integrity if the boron concentration is less than 2000 ppm was not added. No screening criteria apply for this requirement since the boron concentration limit is only a requirement for fuel handling accidents. Since the requirements of ITS LCO 3.6.1 immediately stop all fuel movement in this condition, there is no for containment integrity. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and have been deleted. This is a Ginna TS Category (v.b.50) change..

17. Technical Specification 3.7

- i. TS 3.7.1.1.b, 3.7.1.1.d, and 3.7.1.1.e - The cold shutdown or refueling requirements (MODES 5 and 6) for the 480 V safeguards buses, batteries and DC trains, and 120 VAC instrument buses were revised from requiring only one train to be OPERABLE to require the necessary train(s) to support all other LCO requirements. Consequently, one or both trains of these systems may be required depending on other system requirements (e.g., RHR). In MODES 5 and 6, sufficient electrical power redundancy must be available to mitigate an event coincident with either a loss of offsite power, loss of all onsite standby emergency power, or a worse case single failure. This change ensures that all necessary electrical support systems are OPERABLE to respond to a DBA or a transient. This is a Ginna TS Category (iv.a) change.
- ii. TS 3.7.1.2 - Cold or refueling requirements (MODES 5 and 6) for the DG fuel oil supply and the battery parameters have been added to provide restoration times for specified conditions consistent with the ITS. These times are sufficient to complete restoration of the degraded parameter prior to declaring the component inoperable and is acceptable based on the low probability of an event during this brief period and the fact that the component remains capable of performing most required functions. This is a Ginna TS Category (v.a) change.
- iii. TS 3.7.2.1.b.2, 3.7.2.2.a, and 3.7.2.2.b - The requirements for two offsite sources were not added. The current actions allow the plant to operate indefinitely with one offsite source inoperable. The new ITS format criteria would not specify these requirements in the TS (i.e., require a component for a MODE change but allow the component to remain inoperable indefinitely once the MODE change is complete). Therefore, these requirements are relocated to the TRM. The offsite power sources are further discussed in Reference 32. This is a Ginna TS Category (iii) change.
- iv. TS 3.7.2.2.b.1 - The actions for an inoperable DG have been revised: (1) to eliminate the testing of the OPERABLE DG if, within 24 hours, it can be determined that the OPERABLE DG is not inoperable due to common cause failure, and (2) to require verification of the offsite power circuit to the affected AC distribution train. In addition, the OPERABLE DG must only be tested once during the 7 day allowed outage for the inoperable DG. The revised action for the OPERABLE DG eliminates unnecessary testing during a period in which the plant relies on only one DG. These are Ginna TS Category (iv.b.2) and (v.a) changes.

- v. TS 3.7.2.2.c - The Completion Time for the action to re-energize the 480 V safeguards bus has been revised from 1 hour to 8 hours. The time is consistent with the ITS which assumes not only restoration of the bus but also the associated load centers, motor control centers, and distribution panels which comprise the AC electrical train. This is a Ginna TS Category (v.b.24) change.
- vi. TS 3.7.2.2.d - This was revised to address the scenario with both offsite power and one DG were inoperable. In this condition, no loss of safety function exists since the remaining DG is available to provide power to one ESF train. However, the time in this Condition should be limited due to the potential to lose multiple safety functions if the remaining DG were lost. Therefore, a Completion Time of 12 hours is provided. However, if both offsite power and one DG were inoperable to the same AC electrical train, then the time would be restricted to 8 hours as discussed in Section D, item 17.v above. This is a Ginna TS Category (v.b.54) change.

18. Technical Specification 3.8

- i. TS 3.8 - The applicability was revised from "during refueling operations" to "CORE ALTERATIONS and irradiated fuel assembly movement within containment." This is an equivalent change since refueling operations can only be related to CORE ALTERATIONS and irradiated fuel assembly movement within containment. This is a Ginna TS category (v.c) change.
- ii. TS 3.8.1.b - The refueling or MODE 6 requirement for the containment radiation monitors which provide monitoring for personnel safety was not added. No screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the containment radiation monitors are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.

- iii. TS 3.8.1.c - The requirement describing the specific applicability of the SRMs was revised. The phrase "whenever geometry is being changed" is covered by the new TS definition of MODE 6. The requirement that one SRM be OPERABLE when core geometry "is not being changed" is covered by the Required Action for one inoperable SRM. This would restrict CORE ALTERATION and positive reactivity additions when core geometry is not being changed. Required Actions were also provided when two SRMs become inoperable or when the audible indication is lost. These new actions require verification of boron concentration every 12 hours and ensures the stabilized condition of the reactor core. These are a conservative revisions and Ginna TS Category (v.a) and (iv.a) changes, respectively.
- iv. TS 3.8.1.e - The requirement describing the specific applicability and frequency of the boron concentration sampling was revised. The phrase "immediately before reactor vessel head removal and while loading and unloading fuel from the reactor" is covered by the new TS definition of MODE 6. This would additionally require boron concentration sampling throughout MODE 6. The sampling frequency, however, was also revised to require sampling every 72 hours. These revisions consider the large volume of the refueling canal, RCS, and refueling cavity and are adequate to identify slow changes in boron concentration. Rapid changes in boron concentration, described in UFSAR 15.4.4.2, are detected by the SRM instrumentation required by new TS 3.9.2. This is a conservative revision and a Ginna TS Category (iv.a) change.
- v. TS 3.8.1.f - The requirement for communication with the control room during CORE ALTERATIONS is not added. No screening criteria apply for this requirement since communications is not part of the primary success path assumed in the mitigation of a DBA or transient. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.
- vi. TS 3.8.1.d (footnote *) and TS 3.8.1.g (footnote *) - The requirement that either the preferred or the emergency power source may be inoperable for each residual heat removal loop is not added. This detail is encompassed in the definition of operability described in new TS 1.1 and the electric power requirements contained in Chapter 3.8. This is a Ginna TS Category (i) change.

- vii. TS 3.8.1.c - The requirement to provide SRM audible indication in the containment was not added. No screening criteria apply for this requirement since the monitored parameter (audible indication in containment) is not assumed in the refueling safety analysis. The safety analysis assumes audible indication in the control room which is denoted by new LCO 3.9.2. The audible indication is for personnel safety only. Further, the audible indication is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.
- viii. TS 3.8.1.iii.1 - The isolation devices which are allowed was revised to include "or equivalent." The use of "or equivalent" for isolation of a containment penetration is consistent with NUREG-1431 and the bases for CTS 3.8 which allow the use of a "material which can provide a temporary ventilation barrier, at atmospheric pressure, for the containment penetrations during fuel movement." Therefore, this is a clarification only to the LCO. This is a Ginna TS Category (v.c) change.
- ix. TS 3.8.2 - The requirement to initiate action "to correct the violated conditions" was not added to the new specifications since this is always an option to exit the Condition. That is, the Condition is not exited, even after completion of the Required Actions, unless either the LCO is met, or the MODE of Applicability is exited. This is a Ginna TS Category (v.c) change.
- x. TS 3.8.2 - The requirement to cease "operations which may increase the reactivity of the core" was not added to LCO 3.9.3 (Containment Penetrations) since the basis for isolation of containment is with respect to a fuel handling accident. The reactivity of the core is with respect to a boron dilution event which is adequately addressed by other LCOs. This is a Ginna TS Category (v.b.48) change.

19. Technical Specification 3.9

- i. TS 3.9.1.1 - The requirements for radioactive material released in liquid effluents to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- ii. TS 3.9.1.2 and TS 3.9.2.4 - The requirements for dose or dose commitment to individuals which results from cumulative liquid effluent discharges during normal operation over extended periods and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I, 40 CFR 141, and 40 CFR 190 limits were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, radioactive liquid effluent dose projected value is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iii. TS 3.9.1.3 - The requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix I, were not added. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.



- iv. TS 3.9.2.1 - The requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- v. TS 3.9.2.2.a, TS 3.9.2.2.c, and TS 3.9.2.4 - The requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- vi. TS 3.9.2.2.b, TS 3.9.2.2.c, and TS 3.9.2.4 - The requirements for dose due to radioiodine, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days released with gaseous effluents were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, these gaseous effluents doses are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- vii. TS 3.9.2.3 - The requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- viii. TS 3.9.2.5 and TS 3.9.2.6 - The specific requirements for which limit concentration of oxygen in a gas decay tank and the quantity of radioactivity contained in each waste gas decay tank were not added. The level of detail is relocated to Explosive Gas and Storage Tank Radioactivity Monitoring Program described in new Specification 5.5.11 and a more generic description is provided. This is a Ginna TS Category (iii) change.
- ix. TS 3.9.2.7 - The requirements for the solid radwaste system which processes wet radioactive waste and operates in accordance with 10 CFR Part 50, Appendix A, for effluent control were not added. The operability of the system is not assumed in the analysis of any DBA or transient. Further, radioactive waste is a non-significant risk contributor to core damage frequency and offsite release. Therefore; the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radiological Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

20. Technical Specification 3.10

- i. TS 3.10.1 - This was revised to include specific actions and Completion Times for cases when the shutdown bank insertion limits and the control bank insertion, sequence, and overlap limits are not within the limits specified in the COLR. These actions require verification within 1 hour that the SHUTDOWN MARGIN is within limits and restoring the associated value to within limits within 2 hours or be in MODE 3 within 6 additional hours. These additions were made to ensure that the control banks and the shutdown bank are available as assumed in the safety analyses. This is a Ginna TS Category (iv.a) change.

- ii. TS 3.10.1.1 - This was revised to include a specific action to initiate boration within 15 minutes when the SHUTDOWN MARGIN is not within limits. The addition of this action ensures that SHUTDOWN MARGIN is monitored and quickly restored within limits. This is a Ginna TS Category (iv.a) change.
- iii. TS 3.10.1.1 and Figure 3.10-2 - These were revised to relocate the SHUTDOWN MARGIN requirements and Figure 3.10-2 to the COLR. SHUTDOWN MARGIN can be used in fuel management and as a variable to solve plant specific problems. SHUTDOWN MARGIN impacts a number of analyses (i.e., uncontrolled boron dilution and steamline break) and is sensitive to many core related parameters such as control bank position, core power level, coolant temperature and cycle specific parameters such as fuel burnup, xenon concentration and boron concentration. The inclusion of SHUTDOWN MARGIN in the COLR provides more flexibility in plant operation, in performing the design, and in obtaining good fuel economics particularly for extended cycle operation. With the SHUTDOWN MARGIN included in the COLR, the core design can be finalized after shutdown so that the actual end of cycle burnup is known which is particularly helpful when the actual burnup differs from the projected value. This is a Ginna TS Category (iii) change.
- iv. TS 3.10.1.2 and TS 3.10.1.3 - These were revised to indicate only low power PHYSICS TEST exceptions for the shutdown and control bank insertion limits. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the insertion limits. This is a Ginna TS Category (vi) change.
- v. TS 3.10.1.3 and Figure 3.10-1 - These were revised to relocate the control rod insertion limits and the sequence and overlap limits to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. This is a Ginna TS Category (iii) change.
- vi. TS 3.10.1.5 - This was not added to the new specifications. None of the PHYSICS TESTS currently performed at Ginna Station currently require a relaxation of the SHUTDOWN MARGIN requirements. Therefore none of these SHUTDOWN MARGIN PHYSICS TESTS exceptions or Required Actions are necessary. This is a Ginna TS Category (vi) change.
- vii. TS 3.10.2.2 - This was revised to remove the low power PHYSICS TESTS exception since new LCO 3.2.1 and LCO 3.2.2 which contain the peaking factor requirements are only applicable in MODE 1. This is a Ginna TS Category (v.a) change.

- viii. TS 3.10.2.3 - This was revised to remove the PHYSICS TEST exceptions for the QPTR. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the QPTR limit and the ITS LCO 3.2.4 which contains QPTR is only applicable in MODE 1 with THERMAL POWER \geq 50% RTP. This is a Ginna TS Category (vi) change.
- ix. TS 3.10.2.8, TS 3.10.2.9 and TS 3.10.2.10 - These were revised to remove the PHYSICS TEST exceptions for AFD. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the AFD limits and the ITS LCO 3.2.3 which contains AFD is only applicable in MODE 1 with THERMAL POWER \geq 15% RTP. This is a Ginna TS Category (vi) change.
- x. TS 3.10.3.1.a - This was revised to reduce the minimum T_{avg} for the rod drop test from 540°F to 500°F. The 500°F temperature is conservative since the water will be slightly denser at the lower temperature which has the potential to slow down the dropped rods. This change would enable the plant to complete the rod drop test at an earlier time during plant startup and is consistent with NUREG-1431. This is a Ginna TS Category (v.a) change.
- xi. TS 3.10.4.1 - This was revised to indicate only low power PHYSICS TEST exceptions for control bank alignment. Ginna Station currently does not perform a PHYSICS TEST in MODE 1 which would require the exception of the alignment limits. This is a Ginna TS Category (vi) change.
- xii. TS 3.10.4.2 and TS 3.10.4.3 - These were revised to remove conditions of rod inoperability due to being immovable. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if they are immovable. Reference to full length rods was also removed since there are no part length rods in the reactor core. This is a Ginna TS Category (v.c) change.
- xiii. TS 3.10.4.3.2 - This was revised to remove the requirement to declare a misaligned rod inoperable when the rod cannot be restored to within the alignment limits in 1 hour. The ITS Bases state that the rods are considered to be OPERABLE if they are trippable even if they are immovable. This is a Ginna TS Category (v.c) change.
- xiv. TS 3.10.4.3.2.a - This option for restoring a rod to within alignment was removed from the LCO and relocated to the Bases for ITS 3.1.4 which is controlled under the Bases Control Program. This is a Ginna TS Category (iii) change.

- xv. TS 3.10.4.3.2.b.iii and Table 3.10-1 - These were revised to remove Table 3.10-1 from the specifications. The ITS requires evaluations of accident analysis to be performed to determine that the core limits will not be exceeded during a Design Basis Accident. An evaluation of each of the analyses on Table 3.10-1 may not be required to determine that the core limits will not be exceeded. This table was relocated to the TRM. This is a Ginna TS Category (iii) change.
- xvi. TS 3.10.4.3.2.b and TS 3.10.4.3.2.c - These were revised to remove the requirement to reduce the high neutron flux trip setpoint to $\leq 85\%$ RTP when the power level is reduced to $\leq 75\%$ RTP. This required action is deleted based on agreements between the NRC and the owners groups and is consistent with WCAP-13029 (Ref. 50) which states that the safety analyses results would not be significantly affected by changes to their initial assumptions as a result of increased peaking factors caused by rod misalignment. Additionally, the peaking factor limit verification within 72 hours and the re-evaluation of the safety analysis within 5 days that are required by this specification provide further assurance that the assumptions made in the safety analysis are preserved. This is a Ginna TS Category (v.b.52) change.
- xvii. TS 3.10.4.4 - This was revised to include an action to verify SHUTDOWN MARGIN or initiate boration within 1 hour when more than one rod is out of alignment. The ITS Bases state that 1 hour is a reasonable time based on the time required for potential xenon distribution and the low probability of a accident. This is a Ginna TS Category (v.a) change.
- xviii. TS 3.10.5.1 - This was revised to add an action statement to clarify that if more than one MRPI is inoperable per group for one or more groups or more than one demand position indicator per bank is inoperable for one or more banks then the plant must enter 3.0.3 immediately. This is a Ginna TS Category (v.a) change.
- xix. TS 3.10.5.2.a - This was revised to allow 4 hours (instead of immediately) to verify rod position. The rod position cannot be determined immediately. It takes time to acquire the data and obtain the results. "Immediately" is considered a start time not a completion time. The ITS Bases state that 4 hours provides an acceptable period of time to verify the rod positions while a reduction to $\leq 50\%$ RTP will avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP with 2 or more rods misaligned. This is a Ginna TS Category (v.c) change.

- xx. TS 3.10.2.1 - This was revised to require measurement of the power distribution after each fuel reloading prior to operation of the plant at or above 75% RTP instead of prior to 50% RTP consistent with ITS. This requirement ensures that the design limits are not exceeded when RTP is achieved, since peaking factors are usually decreased as power increases. Requiring this surveillance at 75% versus 50% still provides the necessary margin to ensure that design safety limits are not exceeded and provides the operator with more flexibility during power ascension following a refueling. This is a Ginna TS Category (v.b.25) change.
- xxi. TS 3.10.2.1 - This was revised to delete the requirement to verify QPTR using movable incore detectors or core exit thermocouples with one power range detector inoperable at THERMAL POWER \geq 75% RTP and replaced with a requirement to perform a flux map to verify that hot channel factors are within limits consistent with ITS. The incore detectors are not used to verify QPTR but rather to verify that the core power distribution is acceptable. Ginna Station does not have 8 pairs of symmetric thimble plugs which are necessary to perform a partial flux map and thus would have to complete a full core flux map to verify that the core power distribution is acceptable. This change is consistent with current interpretations at Ginna Station and is preferred by Ginna Station licensed personnel. This is a Ginna TS Category (v.c) change.
- xxii. TS 3.10.2.2 - This was revised to require the hot channel factors be within limit only in MODE 1. The proposed Applicability does not require the F_Q or $F_{\Delta H}$ limits to be met in MODES 2 - 5 or during refueling. As described in the ITS Bases, F_Q and $F_{\Delta H}$ must be within limits during MODE 1; however, such limits are not necessary in MODE 2 because there is insufficient stored energy in the fuel or being transferred to the coolant to require these limits. This is a Ginna TS Category (v.b.26) change.
- xxiii. TS 3.10.2.2 - This was revised to relocate the limits for $F_Q(Z)$ and $F_{\Delta H}$ and the Figure 3.10-3 to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. This is a Ginna TS Category (iii) change.
- xxiv. TS 3.10.2.2 - This was revised to include an administrative Action to reduce the AFD acceptable operational limits specified in the COLR by the percentage that F_Q exceeds the limit. This is necessary since a change in F_Q can adversely impact AFD limits. A Completion Time of 8 hours is allowed to perform this action. This is a Ginna TS Category (iv.a) change.

- xxv. TS 3.10.2.2 - This was revised to allow 72 hours (instead of 24 hours) to reduce the Overpower ΔT and the Overtemperature ΔT trip setpoints when F_Q or $F_{\Delta H}$ is not within limits consistent with NUREG-1431. This section was also revised to include a Completion Time of 72 to reduce the Power Range Neutron Flux High trip setpoints. These actions provide further protection against the consequences of severe transients with unanalyzed power distributions. The 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the initial prompt reduction in THERMAL POWER. This is a Ginna TS Category (v.b.27) change.
- xxvi. TS 3.10.2.2 - This was revised to add a Required Action to be in MODE 2 within 6 hours if the Required Actions and associated Completion Times for the Condition when F_Q or $F_{\Delta H}$ is not within limits is not met. This action places the plant in a condition outside of the Applicability requirements for the Hot Channel Factor requirements. The Completion Time of 6 hours is sufficient to reach MODE 2 from full power operation in an orderly manner without challenging plant systems. This is a Ginna TS Category (iv.a) change.
- xxvii. TS 3.10.2.3 and 3.10.2.4 - These were revised to specifically define the Applicability requirements for QPTR as MODE 1 with THERMAL POWER > 50% RTP. This Applicability is consistent with the current requirements for Ginna Station since continued operation is allowed for an unlimited period of time when THERMAL POWER is < 50% RTP. The ITS Bases state that below 50% RTP there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. In addition, the LCO limit of 1.12 was removed since the primary limit of 1.02 will be reached initially and actions will already be in progress to address the tilt. THERMAL POWER will continue to be reduced if the tilt ratio continues to increase. This revision is consistent with the changes made in WCAP-12159 (Ref. 51) and current industry practice. These are Ginna TS Category (v.a) changes.

- xxviii. TS 3.10.2.3 - This was revised to limit the THERMAL POWER relative to the percentage of quadrant power tilt, (i.e., limit power to 3% below RTP for each 1% by which the QPTR exceeds 1.00) instead of requiring an immediate action to go below 75% RTP. The reduction to 75% RTP essentially employs a 2% RTP reduction for each 1% the QPTR was above 1.00 up until QPTR equalled 1.12 where a reduction to 50% RTP was required. The proposed change would provide flexibility with the initial reduction, but would require at least a 3% RTP reduction for each 1% QPTR exceeded 1.00. Thus, the proposed change while requiring a smaller reduction for small tilts is more conservative for larger tilts which would suggest a more serious problem. This revision is consistent with the changes made in WCAP-12159 (Ref. 51) and current industry practice. The requirement to measure the hot channel factors when the QPTR exceeds 1.02 is changed from within 2 hours to within 24 hours since the THERMAL POWER is appropriately limited within 2 hours. The 24 hour Completion Time takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. This is a Ginna TS Category (v.a) change.
- xxix. TS 3.10.2.4 - This was revised to delete the requirement to identify the cause of the tilt or limit power to < 50% RTP. Identification of the cause of the tilt is not always possible and other actions already underway are adequate to assure safe operation of the plant (e.g., surveillances). This change is consistent with WCAP-12159 (Ref 53). This is a Ginna TS Category (v.b.28) change.
- xxx. The following Required Actions were added for the Condition when QPTR is not within the limit: These are Ginna TS Category (iv.a) changes.
- a. A requirement to verify by calculation that the QPTR is within limits and limit power accordingly every 12 hours.
 - b. A requirement to recalibrate the excore detectors prior to increasing THERMAL POWER above the limit in TS 3.10.2.3. This action is modified by a Note that requires verification that the hot channel factors are within limits prior to recalibration of the excore detectors.

- c. A requirement to verify F_Q and $F_{\Delta H}$ within limits either within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit in TS 3.10.2.3. This action is modified by several Notes. The first Note clarifies that when the QPTR alarm is due to instrumentation alignment this action does not need to be completed. The second note allows this action to be completed only after the excores have been recalibrated. The third note clarifies that the Completion Time applicable first is the one that must be met.
 - d. A requirement to reduce power to < 50% RTP within 4 hours if the initial Required Actions are not met within the associated completion time. This takes the plant out of the Applicability when the actions are not met and provides an additional action before plant shutdown is required.
- xxxi. TS 3.10.2.5 - This was deleted since the 1.12 QPTR limit no longer applies and the Applicability requirement for QPTR has been revised to > 50% RTP. Actions already in progress (i.e., limiting power by 3% below RTP for each 1% QPTR exceeds 1.00) are sufficient to address the tilt. This is a Ginna TS Category (v.c) change.
 - xxxii. TS 3.10.2.7 - This was revised to require a measurement of the target flux difference within 31 EFPD after each refueling instead of within 92 EFPD. This requirement is also modified with a note that allows the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling. The change to within 31 EFPD is more conservative than within 92 EFPD and is necessary to perform the initial monthly target flux difference update also required by TS 3.10.2.7. This is a Ginna TS Category (v.a) change.
 - xxxiii. TS 3.10.2.8 - This was revised to relocate the AFD target band to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The Applicability requirement was also revised to specify MODE 1 with THERMAL POWER > 15% RTP. As described in the ITS Bases, this Applicability is acceptable because of the low amounts of stored or transferred energy in the lower MODES. The AFD at these lower conditions does not affect the consequences of the design basis events. Additionally, the low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. These are Ginna TS Category (iii) and (v.c) changes, respectively.

- xxxiv. TS 3.10.2.9 - This was revised to specify 15 minutes (instead of immediately) to restore AFD to within the target band and then immediately initiate actions to reduce THERMAL POWER to < 90% RTP if the AFD is not restored within the initial 15 minutes. This is consistent with the intent of the current Ginna Station technical specifications. "Immediately" is considered a start time not a completion time and 15 minutes is considered a sufficient amount of time to restore AFD within limits without allowing the plant to remain in an unanalyzed condition for an extended period of time prior to a reduction in power. This is a Ginna TS Category (v.c) change.
- xxxv. TS 3.10.2.10a - This was revised to relocate the AFD target band and the acceptable operation limits to the COLR. This change is consistent with NUREG-1431 and provides flexibility during reload core design. The Applicability requirement was also revised to specify MODE 1 with THERMAL POWER > 15% RTP. As described in the ITS Bases, this Applicability is acceptable because of the low amounts of stored or transferred energy in the lower MODES. The AFD at these lower conditions does not affect the consequences of the design basis events. Additionally, the low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP. These are Ginna TS Category (iii) and (v.c) changes respectively.
- xxxvi. TS 3.10.2.12 - This was revised to require a verification that the AFD is within limits and to log the AFD every 15 minutes with THERMAL POWER \geq 90% RTP and once every hour with THERMAL POWER < 90% RTP when the AFD monitor alarm is inoperable instead of every hour for the first 24 hours and every half hour thereafter. This modification reflects the importance of staying within the target band at above 90% RTP and is consistent with the Required Action if the AFD is outside the target band. This is a Ginna TS Category (v.c) change.



21. Technical Specification 3.11

- i. TS 3.11.1 - This was revised to require that the Auxiliary Building Ventilation System (ABVS) be OPERABLE when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. The specific components which are required for the ABVS to be considered OPERABLE were relocated to the bases similar with the structure of NUREG-1431 and the ITS Writer's Guide. The bases for LCO 3.7.10 now require that one of the two 100% capacity Auxiliary Building main exhaust fans, exhaust fan C, the SFP Charcoal Absorber System, and all associated ductwork, valves and dampers be OPERABLE. In addition, TS 3.11.1.c was revised to require a negative pressure within the Auxiliary Building operating floor with respect to the outside environment instead of requiring all doors, windows, and other direct openings between the operating floor area and the outside to be closed. This change provides consistency with assumptions of the fuel handling accident as described in the bases. This change also provides a much clearer specification which is easier for licensed personnel to read and understand without any reduction in actual requirements. These are Ginna TS Category (i) and (v.a) changes, respectively.
- ii. TS 3.11.2 - The requirement to continuously monitor radiation levels in the SFP area was not added to the new specifications. No screening criteria apply for this requirement because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the SFP radiation levels only provide a backup source to a SFP problem. Other LCOs provide adequate verification of SFP primary indications (i.e., level and boron concentration) which ensure that all accident analysis assumptions are met. Since a fuel handling accident can only occur as a result of fuel movement, personnel would be stationed within the Auxiliary Building and immediately aware of a problem. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- iii. TS 3.11.3 and 3.11.5 - The heavy load restriction for movement of loads over the SFP was not added to the new specifications. No screening criteria apply for this requirement because the heavy load limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This change is consistent with WCAP-11618 (Ref. 52) and is a Ginna TS Category (iii) change.

- iv. TS 3.11.4 - The SFP water temperature limit was not added to the new specifications. No screening criteria apply for this requirement because the SFP water temperature limit of this LCO is not an initial condition of a DBA or transient analysis. The requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
22. Technical Specification 3.12
- i. TS 3.12.1 - The requirement for the number of thimbles per quadrant required to OPERABLE during recalibration of the excore axial off-set detection system was not added to the new specifications. The requirements for this surveillance are not an initial assumption of any DBA or transient analysis. Therefore, this specification does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
23. Technical Specification 3.13
- i. TS 3.13 - The requirements for snubbers operability were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to the Inservice Inspection Program. This is a Ginna TS Category (iii) change.
24. Technical Specification 3.14
- None.
25. Technical Specification 3.15
- i. TS 3.15.1 - The LTOP exception during secondary side hydrostatic testing was relocated as a NOTE to new LCO 3.4.12. This is a Ginna TS Category (v.c) change.
 - ii. TS 3.15.1 - The PORV setpoint during LTOP conditions was relocated to the PTLR consistent with LCO 3.4.12. This is a Ginna TS Category (iii) change.
 - iii. TS 3.15.1 - The accumulators are now required to be isolated when the accumulator pressure is greater than the maximum RCS pressure for existing cold leg temperatures as specified in the PTLR consistent with Condition C of LCO 3.4.12. This new requirement prevents an accumulator from overpressurizing the RCS and causing an actuation of the LTOP System. The operator is instructed to isolate or depressurize the affected accumulator under these conditions. This is a Ginna TS Category (iv.a) change.

- iv. TS 3.15.1.1 - A new requirement was added when a PORV is inoperable during MODES 5 and 6 due to the increased consequences from an overpressurization event under these conditions. This new requirement specifies that the PORV must be restored to OPERABLE status within 72 hours. The limit of 72 hours with one PORV inoperable is consistent with the allowed outage time for one train of ECCS equipment during MODES 1, 2 and 3. This is a Ginna TS Category (iv.a) change.
- v. TS 3.15.1.3 - The reporting requirement for the low temperature overpressure protection (LTOP) system operation was revised. The reporting requirement to include documentation of all challenges to the pressurizer power operated relief valves is detailed in proposed TS 5.6.4, "Monthly Operating Reports" and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. This is a Ginna TS Category (i) change.
- vi. TS 3.15 - The Applicability was revised to specify that LTOP is only required in MODES 5 and 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manway are closed and secured in position. This change is consistent with the current requirements for isolating the SI pumps for LTOP conditions (3.3.1.7 and 3.3.1.8) and the ITS such that there is no real change in the MODE of Applicability. This is a Ginna TS Category (v.c) change.

26. Technical Specification 3.16

- i. TS 3.16.1 and Table 3.16-1 - The requirements for the radiological environmental program which provides measurements of radiation and of radioactive materials in those exposure pathways and for specified radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- ii. TS 3.16.2 - The requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- iii. TS 3.16.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

27. Technical Specification 4.0

- i. A new section SR 3.0.1 was added which establishes the requirements and limitations that the SRs must meet during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with SR 3.0.1. This SR provides clarifying and descriptive information for the SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.a) change.

- ii. TS 4.0 - This was revised to clarify the basic application of the 25% extension to routine surveillances consistent with the use and format of the ITS. The interval extension concept is based on scheduling flexibility for repetitive performances. There are clarifications provided in SR 3.0.2 for Surveillances which are not repetitive in nature and essentially have no interval as measured from the previous performance. This precludes the ability to extend these performances. The existing Specification 4.0 can be interpreted to allow the extension to apply to all Surveillances. An additional clarification provides the basis for consistency in scheduling flexibility for all performances of periodic requirements, whether they are Surveillances or Required Actions. The intent remains to perform the activity, on the average, once during each specified interval. This section does not provide any new requirements but provides clarification and a description of SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.c) change.

- iii. A new section SR 3.0.3 was added which establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. The SR permits the declaration of the LCO-not-met to be delayed for up to 24 hours or up to the limit of the specified Frequency (whichever is less), and eliminates confusion in applying the correct ACTION time limits at the end of this 24 hour period. The vast majority of surveillances performed demonstrate that systems or components, in fact, are OPERABLE. When a Surveillance is missed, it is primarily a question of OPERABILITY that has not been verified by the performance of the required surveillance. Based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance, the NRC has concluded that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times of the ACTIONS are less than the 24 hour limit or a shutdown is required to comply with ACTIONS (Ref. 53). This section, in part, provides new requirements consistent with the use and format of the ITS. This is a Ginna TS Category (iv.a) change.

- iv. A new section SR 3.0.4 was added which establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability. This section does not provide any new requirements. Previous guidance provided by the NRC (e.g., Generic Letter 87-09) regarding the intent and interpretation of existing Specifications is consistent with SR 3.0.4. This SR provides clarifying and descriptive information for the SRs applicability consistent with the use and format of the ITS. This is a Ginna TS Category (v.a) change.

28. Technical Specification 4.1

- i. The following changes were made to TS 4.1.1 or Table 4.1-1:
 - a. Table 4.1-1, Columns 2 (Calibrate) and 3 (Test) - Various calibration and testing interval requirements for RTS and ESFAS Functions were revised consistent with NUREG-1431. Changes to the testing interval requirements different from those identified and discussed in NUREG-1431 are discussed with the specific changes to the Functional Units. This is a Ginna TS Category (v.b.15) change.
 - b. The following new requirements were added to Table 4.1-1 (Ginna TS Category (iv.a) changes):
 - 1. SR 3.4.2.2 - requires verification every 30 minutes that T_{avg} for each RCS loop is $> 540^{\circ}\text{F}$ when any RCS loop T_{avg} is known to be $< 547^{\circ}\text{F}$. This surveillance is intended to ensure that the minimum temperature for criticality is not exceeded when the RCS is at less than Hot Zero Power conditions (i.e., 547°F). The surveillance is not required to be performed if the low T_{avg} alarm in each loop is reset with a setpoint $> 540^{\circ}\text{F}$.
 - 2. SR 3.4.3.1 - requires verification every 30 minutes that RCS pressure, temperature, heatup and cooldown rates are within limits. This surveillance is only required during RCS heatup and cooldown operations, and inservice leak and hydrostatic testing. The 30 minute Frequency is based on the fact that heatup and cooldown rates are specified in hourly increments which provides adequate margin to correct minor deviations.

3. SR 3.4.1.1 - requires verification every 12 hours that pressurizer pressure is within limits during MODE 1. This surveillance is similar to current Ginna TS Table 4.1-1, #7 which is performed to support reactor trip functions.
4. SR 3.4.1.2 - requires verification every 12 hours that RCS average temperature is within limits during MODE 1. This surveillance is similar to current Ginna TS Table 4.1-1, #33 which is performed to support reactor trip functions.
5. SR 3.4.1.3 - requires performance of a precision heat balance to verify that RCS flow is within limits every 24 months. This surveillance is required to be performed within 7 days of entering MODE 1 and reaching 95% RTP.
6. SR 3.1.6.1 - Requires verification within 4 hours prior to criticality that the critical control bank position is within limits in the COLR.
7. SR 3.1.6.4 - Requires verification every 12 hours when critical that the sequence and overlap limits for the control banks not fully withdrawn are within limits specified in the COLR.
8. SR 3.1.8.3 - Requires verification every 30 minutes during MODE 2 PHYSICS TESTS that THERMAL POWER \leq 5% RTP. Verification of the THERMAL POWER level will ensure that the initial conditions of the safety analyses are not violated.
9. SR 3.2.4.1 - Verification with a calculation using the power range channels every 7 days that the QPTR is within limits.
10. SR 3.4.2.1 - requires verification within 30 minutes prior to achieving criticality that T_{avg} for each RCS loop is $> 540^{\circ}F$. This surveillance is intended to ensure that the minimum temperature for criticality is not exceeded just prior to achieving criticality.

- c. Table 4.1-1, Functional Units #1, #2, #3, #8, #17, #23, #25, #38a, #38b, #39, #40, #41a, and #41b - The notes or remarks which describe an operational detail, were not added. These details were relocated to the bases or are described in the UFSAR. This is a Ginna TS Category (iii) change.
- d. LCO 3.3.1, Table 3.3.1-1, Function #10 was added for the RCP Breaker Position. This function anticipates the Reactor Coolant Flow - Low trips by monitoring each RCP breaker position to avoid RCS heatup that would occur before the low flow trip actuates. The function ensures that protection is provided against violating the DNBR limit due to loss of flow in either a single loop or two loop configuration. This is a Ginna TS Category (iv.a) change.
- e. LCO 3.3.1, Table 3.3.1-1, Function #14 was added for the SI Input from ESFAS. This function ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. A reactor trip is initiated every time an SI signal is present. This is a Ginna TS Category (iv.a) change.
- f. SR 3.3.1.14, SR 3.3.1.15, SR 3.3.1.16, SR 3.3.1.17, SR 3.3.1.18 were added for the Reactor Trip System Interlocks (P-6 through P-10). These surveillances are provided to ensure reactor trips are in the correct configuration for the current plant status. They are provided to back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Functions are not bypassed. This is a Ginna TS Category (iv.a) change.
- g. Table 4.1-1, Functions #34 and #35 - The requirements for the chlorine gas and ammonia gas instrumentation monitors for control room habitability were not added. No screening criteria apply for these requirements since the monitored parameters are not part of the primary success path in the mitigation of a DBA or transient. These monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the TRM. This is a Ginna TS Category (iii) change.

- h. Table 4.1-1, Functional Units #1 and 2 were revised to require a CHANNEL OPERATIONAL TEST (COT) on the power range and the intermediate range channels within 7 days prior to reactor criticality. The ITS Bases states that the 7 day time limits is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating the PHYSICS TESTS. This is a Ginna TS Category (iv.a) change.
- i. Table 4.1-1, Functional Unit #4 was revised to include a note requiring a channel check every 30 minutes while implementing MODE 2 PHYSICS TEST exceptions. Verification of the RCS temperature will ensure that the initial conditions of the safety analyses are not violated. This is a Ginna TS Category (iv.a) change.
- j. Table 4.1-1, Functional Units #18, #28, and #29 - The Surveillance requirements for radiation monitors R-1 through R-9 and R-17, emergency plan radiation instruments, and environmental monitors, were not added to the new specifications. These process variables are not an initial condition of a DBA or transient analysis. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- k. Not used.
- l. Table 4.1-1, Functional Unit #3 - This was revised to add a requirement which establishes a surveillance for a SRM CHANNEL CALIBRATION in MODE 6. This calibration consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to baseline data and is consistent with current Ginna Station procedures. This is a Ginna TS Category (iv.a) change.
- m. Table 4.1-1, Functional Units #14, #16, and #19 were relocated to the TRM for the same reasons as described in Section D, items 12.i through 12.iv. These are Ginna TS Category (iii) changes.

- ii. The following changes were made to TS 4.1.2 or Table 4.1-2:
- a. Table 4.1-2, #6a was revised to extend the surveillance Frequency of the control rod exercises from monthly to every 92 days. The ITS Bases states that the 92 day Frequency takes into consideration the other information available to the operator in the control room and the channel check which is performed more frequently and adds to the determination of rod operability. This is a Ginna TS Category (v.b.29) change.
 - b. Table 4.1-2, #5 and #6b were revised to remove reference to once every 18 months or each refueling shutdown from the Frequency. These surveillances are only performed during a plant outage or during plant startup, prior to reactor criticality after each removal of the reactor head. This is a Ginna TS Category (v.c) change.
 - c. Table 4.1-2, Functional Unit #7 was revised to relocate the surveillance Frequency of the pressurizer safety valves to the IST Program consistent with SR 3.4.10.1. The Frequency continues to remain in a program requiring NRC approval. This is a Ginna TS Category (iii) change.
 - d. Table 4.1-2, Functional Unit #10 was not added to the new specifications. The requirement for verifying the refueling system interlocks is not an initial condition of a DBA or transient analysis. This requirement does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This change is consistent with WCAP-11618 (Ref. 52) and is a Ginna TS Category (iii) change.
 - e. Table 4.1-2, Functional Unit #13 was revised per SR 3.6.6.8 to require verification of the spray additive tank NaOH concentration once every 184 days instead of monthly. This change is acceptable since the spray additive tank is normally maintained isolated at power such that changes to the NaOH concentration or level are not expected. This is a Ginna TS Category (v.b.30) change.

- f. Table 4.1-2, Functional Unit #15 was revised to require RCS water inventory balances every 72 hours during steady state operation versus daily consistent with SR 3.4.13.1. This increased surveillance interval is considered acceptable based on the leakage detection systems required to be OPERABLE by LCO 3.4.15 and the various indications available to operators (e.g., volume control tank level and radiation alarms). This is a Ginna TS Category (v.b.31) change.
- g. Table 4.1-2, Functional Unit #17 was revised to only require verification of SFP boron concentration once every 31 days when fuel is stored in the SFP and the position of fuel assemblies which were moved in the SFP have not been verified. The current monthly requirement (regardless of the status of the SFP verification) is not reflected in the fuel handling accident analysis which does not credit the availability of soluble boron. This is a Ginna TS Category (v.b.32) change.
- h. Table 4.1-2, Functional Unit #19 - The trip function requirement for the Circulation Water Flood Protection was not added. The Circulation Water Flood Protection instruments only provide an anticipatory turbine trip and is not assumed in the Ginna Station safety analysis. These instruments do not monitor parameters which are initial assumptions for a DBA or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design basis event. Therefore, the requirement specified for this function does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- i. The following new requirements were added to Table 4.1-2 (Ginna TS Category (iv.a) changes):
 - 1. SR 3.1.1.1 - Requires verification every 24 hours that the SHUTDOWN MARGIN is within the limits. The ITS Bases state that a Frequency of every 24 hours is based on the generally slow change in boron concentration and the low probability of an event occurring without the required SDM.

2. SR 3.1.3.1 - Requires verification prior to entering MODE 1 after each refueling that MTC is within the upper limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.
3. SR 3.1.3.2 - Requires verification prior to entering MODE 1 after each refueling that MTC will be within the 70% RTP MTC upper limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.
4. SR 3.1.8.3 - Require verification every 24 hours while implementing the MODE 2 PHYSICS TESTS exceptions that the SHUTDOWN MARGIN is within the limits. The ITS Bases state that a Frequency of every 24 hours is based on the generally slow change in boron concentration and the low probability of an event occurring without the required SDM.
5. SR 3.5.1.1 - requires verification every 12 hours that each accumulator motor-operated isolation valve is fully open above 1600 psig.
6. SR 3.5.1.3 - requires verification every 12 hours of an upper limit for the nitrogen pressure blanket in the accumulators to prevent lifting of the relief valve and overpressurization of the tank. A value of 790 psig was selected since it is above the accumulator pressure upper alarm setpoint of 760 psig and below the relief valve setpoint of 800 psig.
7. SR 3.5.1.4 - requires verification every 31 days on an STAGGERED TEST BASIS of an upper limit for boron concentration in the accumulator since this limit is used in determining the time frame which boron precipitation is addressed post LOCA. The value specified in the COLR was selected since this would not create the potential for boron precipitation in the accumulator assuming a containment (and accumulator) temperature of 60°F. This is also bounded by the containment sump pH calculations and assumptions used for chemical spray effects.

8. SR 3.5.1.5 - requires verification every 31 days that power is removed from the accumulator isolation valve operator above 1600 psig. This surveillance is consistent with current TS 3.3.1.1.i. A value of 1600 psig was selected (i.e., the same value as that for accumulator operability) since the RCS pressure interlock (i.e., P-11) as discussed in NUREG-1431 does not exist at Ginna Station. Therefore, there is no interlock signal to open the isolation valves in the event that they are closed.
9. SR 3.5.4.2 - requires verification every 7 days of an upper limit for boron concentration in the RWST since this limit is used in determining the time frame which boron precipitation is addressed post LOCA. The value specified in the COLR was selected since this would not create the potential for boron precipitation in the RWST assuming an Auxiliary Building (and RWST) temperature of 50°F. This is also bounded by the containment sump pH calculations and assumptions used for chemical spray effects.
10. SR 3.6.5.1 - requires verification every 12 hours that containment average air temperature is $\leq 120^{\circ}\text{F}$.
11. SR 3.6.6.8 - requires verification every 184 days that the spray additive tank volume is ≥ 4500 gallons.
12. SR 3.7.11.1 - requires verification every 7 days that ≥ 23 feet of water is available above the top of the irradiated fuel assemblies seated in the storage racks during fuel movement in the SFP. This verification is required since the fuel handling accident assumes that at least 23 feet of water is available with respect to iodine releases.
13. SR 3.7.13.1 and SR 3.7.13.2 - verification prior to fuel movement in the SFP that the associated fuel assembly meets the necessary requirements for storage in the intended region (e.g, enrichment limit, burnable poisons present). This verification is required to limit the amount of time that a fuel assembly could be misloaded in the SFP.

14. SR 3.7.6.1 - requires verification every 12 hours that the CST volume is $\geq 22,500$ gallons. This ensures that the minimum volume of condensate is available for the preferred AFW System following an accident.
15. SR 3.7.7.1 - requires verification every 31 days that each CCW manual and power operated valve in the CCW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the CCW System is capable of performing its function following a DBA to provide cooling water to safety related components.
16. SR 3.7.7.2 - requires performance of a complete cycle of each CCW motor operated isolation valve to the RHR heat exchangers in accordance with the IST Program. This ensures that the normally closed motor operated valves are capable of being opened following a DBA.
17. SR 3.7.8.2 - requires verification every 31 days that each SW manual and power operated valve in the SW pump train or loop header flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. This Surveillance ensures that the SW System is capable of performing its function following a DBA to provide cooling water to safety related components.
18. SR 3.9.4.1 and 3.9.5.1 - requires verification every 12 hours in MODE 6 that one RHR loop is in operation and circulating reactor coolant. This ensures that the RCS is being mixed as assumed for boron dilution events and that decay heat removal continues during shutdown.
19. SR 3.9.3.1 - requires verification every 7 days that all containment penetrations which communicate to the outside environment are in their required state in MODE 6. This ensures that containment is in the correct state prior to and during fuel movement.
20. SR 3.6.3.1 - requires verification every 31 days that the mini-purge valves are closed, except when the penetration flow path is being used under administrative control. This ensures that the flow paths which provide a direct path from containment to the outside environment are in the correct position.

21. SR 3.7.8.1 - requires verification every 24 hours that the greenhouse bay water level and temperature are within limits. This ensures that the ultimate heat sink source is within the assumptions of the accident analyses.
22. SR 3.7.8.3 - requires verification every 31 days that all SW loop header cross-tie valves are in the correct position. This ensures that the valves are either locked opened or closed as necessary to support the accident analyses.
23. SR 3.5.2.7 - requires a visual verification every 24 months that the RHR containment sump suction inlet line is not obstructed and that the screen shows no evidence of structural distress or abnormal corrosion. This ensures that the RHR system will not become plugged by expected debris which may exist in containment post-LOCA.
24. SR 3.1.3.3 - requires verification prior to entering MODE 1 after each refueling that MTC will be within the EOL lower MTC limit. The ITS Bases state that meeting the limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.
25. SR 3.2.3.1 - requires verification every 12 hours that the AFD monitor is OPERABLE. This ensures that the AFD monitor is available to detect changes in AFD and provide necessary indication to operators.
26. SR 3.2.4.1 - requires verification every 12 hours that the QPTR monitor is OPERABLE. This ensures that the QPTR monitor is available to detect changes in QPTR and provide necessary indication to operators.
27. SR 3.6.6.5 - requires verification every 31 days that cooling water is flowing through each CRFC unit. This ensures that the heat removal capability of the CRFC units is verified to be available as assumed in the accident analyses.
28. SR 3.5.2.3 - requires verification every 31 days that each breaker or key switch is in the correct position of valves required to be depowered or powered. This ensures that no single active failure will fail both ECCS trains.

- j. Table 4.1-2, Functional Units #1 and #2 - These were not added to the new specifications for the reasons discussed in Section D, item 11.i. This is a Ginna TS Category (iii) change.
 - k. Table 4.1-2, Functional Unit #16 - This was revised to only require a verification of DG fuel oil inventory once every 31 days instead of daily. Since the storage tanks are of passive design and are provided with various level alarms, verification every 31 days is considered adequate. This is a Ginna TS Category (v.b.33) change.
 - l. Table 4.1-2, Functional Unit #4 - This was relocated to the TRM for the same reasons as described in Section D, item 12.iv. This is a Ginna TS Category (iii) change.
 - m. Table 4.1-2, Functional Unit #12. - This was relocated to the IST Program since it does not meet any of the requirements for inclusion in the ITS. This is a Ginna TS Category (iii) change.
 - n. Table 4.1-2, Functional Unit #18 - The Frequency for determining gross specific activity of the secondary system was revised from once every 72 hours to once every 31 days. In addition, the determination of I-131 was also changed to once every 31 days independent of the last activity level since the current Ginna TS allow up to 6 months between tests. These changes are all consistent with NUREG-1431. This is a Ginna TS Category (v.b.51) change.
- iii. The following changes were made to TS 4.1.3 or Table 4.1-3:
- a. Table 4.1-3 - The Post Accident Monitoring Instrumentation Functions required by this specification were revised to include only Regulatory Guide 1.97, Type A and Category 1 variables. These Functions are denoted in UFSAR Table 7.5-1 and have been previously reviewed and approved by the NRC (Ref. 35). This is a Ginna TS Category (v.c) change.

iv. The following changes were made to TS 4.1.2 or Table 4.1-4:

- a. Table 4.1-4, Functional Unit #1 was revised per SR 3.4.16.1 to only require verification of reactor coolant gross specific activity once every 7 days when $T_{avg} \geq 500^{\circ}\text{F}$ versus once every 72 hours above Cold Shutdown (i.e., $T_{avg} \geq 200^{\circ}\text{F}$). The increased surveillance interval is acceptable based on the small probability of a gross fuel failure during the additional 4 days. Fuel failures are more likely to occur during startup or fast power changes and not during steady state power operation during which the majority of sampling is performed. Gross fuel failures will also result in Letdown radiation alarms and possibly containment radiation alarms providing additional operator indication. Only requiring this surveillance when $T_{avg} \geq 500^{\circ}\text{F}$ provides consistency with the LCO Applicability. This is a Ginna TS Category (v.b.34) change.
- b. Table 4.1-4, Functional Unit #2 was revised per SR 3.4.16.2 to require verification of DOSE EQUIVALENT I-131 when $T_{avg} \geq 500^{\circ}\text{F}$ instead of above 5% reactor power. This conservative change provides consistency with the LCO Applicability. This is a Ginna TS Category (v.a) change.
- c. Table 4.1-4, Functional Unit #3 was revised per SR 3.4.16.3 to delay determination of E until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation following the reactor being subcritical for ≥ 48 hours. The 31 days was added to ensure that radioactive materials are at equilibrium in order to provide a true representative sample for E determination and eliminate possible false samples. This is a Ginna TS Category (v.b.53) change.

v. The following changes were made to TS 4.1.4 or Table 4.1-5:

- a. Table 4.1-5, Functional Unit #3b was revised to require a channel check of particulate sampler R-11 every 12 hours versus weekly. This is required since R-11 is being used to monitor RCS leakage and may be the only installed system OPERABLE to perform this task for up to 30 days per new LCO 3.4.15.

- b. TS 4.1.4 and Table 4.1-5 - The Radioactive Effluent Monitoring Instrument Functions required by this specification were not added to the new specifications since these process variables are not an initial condition or a DBA or transient analysis. Therefore, the requirements specified for these functions do not satisfy the NRC Final Policy Statement technical specification screening criteria and were relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- c. TS 4.1-5, Functional Unit #3a and #3b were revised to only require the functional test of the valves actuated by R-11 and R-12 once every 24 months versus quarterly. This change is consistent with NUREG-1431 and is considered acceptable since these channels are redundant to the containment isolation signal. As such the accident analysis do not take specific credit for R-11 and R-12 to isolate the containment purge valves. Also, a functional test of the channels (minus actuation of the valves) is to be done quarterly. This is a Ginna TS Category (v.b.49) change.
- d. The CHANNEL CHECK of R-11 was revised from weekly to daily in MODES 1, 2, 3, and 4, and during MODE 6 when required by LCO 3.9.3. This is a conservative change which requires R-11 and r-12 to be checked at the same frequency. This is a Ginna TS Category (v.a) change.
- e. Table Note 5 was deleted since requiring that the CHANNEL CALIBRATION be traced back to the National Bureau of Standards is not a necessary level of detail to be contained in TS. NUREG-1431 does not contain this level of detail since it does not meet any of the four criteria. Therefore, this Note is relocated to plant procedures. This is a Ginna TS Category (iii) change.

29. Technical Specification 4.2

- i. TS 4.2.1 - The specific requirements for the Inservice Inspection Program, which include Quality Groups A, B, and C components, high energy piping outside of containment, snubbers and steam generator tubes, were not added. The level of detail is relocated to licensee controlled documents (Ginna Station QA Manual, Appendix B) and a more generic description is provided. This is a Ginna TS Category (iii) change.

- ii. TS 4.2.1 - The title of the "Ginna Station QA Manual" was changed to "Nuclear Policy Manual" since this is the location of the IST Program per Reference 64. This is a Ginna TS Category (vi) change.

30. Technical Specification 4.3

- i. TS 4.3.5.6 - This surveillance was not added for the reasons discussed under Section C, item 6.xi. This Surveillance has been relocated to the TRM. This is a Ginna TS Category (iii) change.
- ii. TS 4.3.5.3.b - This surveillance was not added since performance of pump testing in accordance with the Inservice Testing program should not be required for an operating RHR pump. The status of a non-operating RHR pump is assured by new SR 3.4.6.3 which requires the verification of the breaker alignment and indicated power available to the pump. The Inservice Testing program testing is mainly performed to ensure adequate performance during accident conditions which far exceeds the requirements during shutdown conditions. This test is not necessary to ensure operability during MODE 4 operations. However, this Surveillance is required for ECCS during MODE 4 (see new SR 3.5.3.1) This is a Ginna TS Category (v.c) change.
- iii. TS 4.3 - The following new requirements were added (Ginna TS Category (iv.a) change):
 - a. SR 3.4.6.3, 3.4.7.3 and 3.4.8.2 - Requires the verification of correct breaker alignment for the non-operating, but required, RHR pump in MODES 4 and 5.
 - b. SR 3.4.9.2 - Requires verification that the total capacity of the pressurizer heaters is ≥ 100 KW once every 92 days.
 - c. SR 3.4.11.2 - Requires a complete cycle of each PORV using the nitrogen system once every 24 months.

- iv. TS 4.3.3.1, 4.3.3.2, and 4.3.3.3 - The requirement that the leakage tests be performed with a minimum test differential pressure of 150 psid was not added to the new specifications. The bases for new LCO 3.4.14 reference ASME, Section XI (Ref. 53) which provides acceptable guidance for performing these leakage tests. This includes adjusting the observed leakage rates for tests that are not conducted at the maximum differential pressure by assuming that leakage is directly proportional to the pressure differential to the one half power. This is a conservative change in most cases since it requires that the PIVs be tested under the maximum differential pressure conditions. This is a Ginna TS Category (v.c) change.
- v. TS 4.3.3.4 - The allowed leakage rates for PIVs was adjusted from a single value for all valves to a value based on valve size consistent with SR 3.4.14.1 and SR 3.4.14.2. This change provides greater information of valve degradation and removes an unjustified penalty on larger valves (Ref. 54). This is a Ginna TS Category (v.c) change.
- vi. TS 4.3.5.5 - This surveillance was not added during MODE 1 operation since there is a reactor trip function which protects the SG level. This is a Ginna TS Category (i) change.
- vii. TS 4.3.1.1 - This requirement was not added to the new specifications since it only states that the reactor vessel must be tested in accordance with 10 CFR 50, Appendix H. Since this requirement is already specified in the CFR, it does not have to be retained with the TS and was deleted. This is a Ginna TS Category (ii) change.
- viii. TS 4.3.3.1 - This was modified to remove the requirement to test the SI cold leg injection and RHR RCS PIVs each cold shutdown. At Ginna Station, these flowpaths are only used for emergency injection (i.e., they are not relied upon or used during cold shutdown conditions). Since the valves are maintained closed at all times, requiring a leak test within 24 hours of being opened or having maintenance performed, and once every 24 months provides adequate protection. A leakage test every 24 months is also consistent with NRC approved OMa-1988. This is a Ginna TS Category (v.b.35) change.
- ix. TS 4.3.4.2 - This was revised to limit the exclusion for testing of the PORV block valves from when "the valve is closed," to "when the valve is closed due to PORV leakage > 10 gpm." This ensures that the block valve is tested under all conditions except those that could potentially result in a plant transient. This is a conservative change. This is a Ginna TS Category (v.a) change.

31. Technical Specification 4.4

- i. TS 4.4.4 - The requirements for the tendon stress surveillances were not added. The level of detail is relocated to the Pre-stressed Concrete Containment Tendon Surveillance Program described in new Specification 5.5.6 and a more generic program description is provided. This is a Ginna TS Category (iii) change.
- ii. TS 4.4.3 - The requirements for the testing of the portion of the RHR system in the recirculation configuration were not added. The level of detail is relocated to the Primary Coolant Sources Outside Containment Program described in new Specification 5.5.2 and a more generic program description is provided. This is a Ginna TS Category (iii) change.
- iii. TS 4.4.1 (except definition for L_a), 4.4.2.1, 4.4.2.2, and 4.4.2.4 - These were not added to the new specifications since this information is contained in 10 CFR 50, Appendix J and does not need to be retained within technical specifications. SRs 3.6.1.1 and 3.6.1.2 provide for the necessary relation from technical specifications to Appendix J (see also Reference 63). These are Ginna TS Category (ii) changes.
- iv. TS 4.4.2.3.a and 4.4.2.3.b - These were revised to require that if the allowed 10 CFR 50, Appendix J leakage limits are exceeded, they must be restored within 1 hour versus 48 hours consistent with LCO 3.6.1. However, the leakage limit of $< 0.6 L_a$ was revised to be consistent with the new Appendix J rule and implementation guidance (i.e., the leakage limit is $< 0.6 L_a$ on a maximum pathway leakage rate basis prior to entering MODE 4 for the first time following each refueling outage and $< 0.6 L_a$ on a minimum pathway leakage rate basis for all other time periods) (see also Reference 63). This is a Ginna TS Category (v.a) change.
- v. TS 4.4.2.4.c - A specified air lock leakage acceptance criteria of $\leq 0.05L_a$ when tested at $\geq P_a$ was added to the new specifications. This acceptance criteria is required to be retained within technical specifications by 10 CFR 50, Appendix J, Section III.D.2(iv) and is consistent with NUREG-1431 and current testing requirements. In addition, a new Surveillance was added to verify that only one door in each airlock can be opened at a time once every 24 months. This test is necessary to ensure that the OPERABILITY of the airlocks, as defined in the new bases for LCO 3.6.2 is maintained. These are Ginna Category (iv.a) changes.

- vi. TS 4.4.2.3.c - The requirement to perform an engineering evaluation if the mini-purge supply and exhaust lines isolation valve leakage exceeds 0.05 L_g was revised to require isolation of the affected penetration within 24 hours. In addition, the affected penetration must be verified isolated once every 31 days if it is outside containment, or once every 92 days if it is inside containment. These changes provide direct guidance to operators which are consistent with NUREG-1431. This is a Ginna TS Category (v.c) change.
- vii. TS 4.4.5.1 - Two new surveillances (SR 3.6.3.2 and SR 3.6.3.3) were added which require verification of the correct position of containment isolation barriers located outside containment once every 92 days and inside containment prior to entering MODE 4 from MODE 5 if it has not been performed within the previous 92 days. These surveillances ensure that the containment isolation barriers remain OPERABLE above MODE 5. These are Ginna TS Category (iv.a) changes.
- viii. TS 4.4.6.2 - The Surveillance Frequency for automatic containment isolation valves has been revised from 18 to 24 months (see Section D, item 1.xii). The response times for CIVs is discussed in the bases for new LCO 3.6.3. This is a Ginna TS Category (v.b.1) change.
- ix. TS 4.4 - Two new Surveillances were added with respect to the hydrogen recombiners (SR 3.6.7.1 and SR 3.6.7.2). The first new Surveillance requires a functional check of the hydrogen recombiners once every 24 months. The second new Surveillance requires that a CHANNEL CALIBRATION be performed on the hydrogen recombiner actuation and control channels once every 24 months. The performance of these SRs ensures that the hydrogen recombiners are OPERABLE and capable of performing their post-accident function. These are Ginna TS Category (iv.a) changes.
- x. TS 4.4.7 - The Frequency for performance of a CHANNEL CHECK of the hydrogen monitors was revised from daily to monthly. In addition, the Frequency for CHANNEL CALIBRATIONS was revised from quarterly to every 24 months. These changes are consistent with NUREG-1431 and are justified by industry experience. These are Ginna TS Category (v.b.21) changes.

32. Technical Specification 4.5

- i. TS 4.5.1.1.a - This was revised to delete the statement that the SI and RHR pumps are prevented from starting during this test. Since these components have recirculation lines available, this statement is not required. This is a Ginna TS Category (v.c) change.
- ii. TS 4.5.2.1 - This was revised to relocate all SI, RHR, and CS pump testing frequencies and discharge pressure requirements to the Inservice Testing program described in new Specification 5.5.8 consistent with the ITS. These are Ginna TS Category (iii) changes, respectively.
- iii. TS 4.5.2.2.c - The test related to accumulator check valve testing for operability every refueling shutdown was relocated to the Ginna Station Inservice Testing program. The valves are currently partially stroke tested quarterly and refurbished every six years. Leakage associated with these check valves is addressed by SR 3.5.1.2. This is a Ginna TS Category (iii) change.
- iv. The following new ITS testing requirements were added (Ginna TS Category (iv.a) change):
 - a. SR 3.5.2.1 - requires verification every 12 hours that ECCS related isolation valves are in their required position. These valves are currently specified in TS 3.3.1.1.g, 3.3.1.1.i, and 3.3.1.1.j.
 - b. SR 3.5.2.2 - requires verification every 31 days that ECCS related valves which are not locked, sealed, or otherwise secured in position are in their correct position.
- v. TS 4.5.2.3 - The requirements denoting the Frequency and conditions of the air filtration system tests were not added to the new specifications. This level of detail is relocated to the Ventilation Filter Testing Program described in new Specification 5.5.10. In addition, the remaining requirements were all relocated to the Administrative Controls section. These are Ginna TS Category (iii) and (i) changes, respectively.
- vi. TS 4.5.2.3.6.a - These test requirements were revised to clarify that two separate tests are performed. A HEPA filter test and a charcoal adsorber bank test are separately performed with each requiring a limit of less than 3 inches of water. This is essentially equivalent to a combined test of less than 6 inches of water and is consistent with specified testing standards. This is a Ginna TS Category (vi) change.

- vii. TS 4.5.1.2 - Two new Surveillances (SR 3.6.6.1 and SR 3.6.6.2) were added to verify the correct position of each manual, power operated, and automatic valve in the NaOH and CS flowpath that is not locked, sealed, or otherwise secured in position. This Surveillance ensures that the NaOH and CS Systems are OPERABLE in accordance with the LCO. These are Ginna TS Category (iv.a) changes.
- viii. TS 4.5.1.2.b - The Frequency of performing the spray nozzle gas test was revised from once every 5 years to once every 10 years consistent with SR 3.6.6.14. The increased surveillance interval is considered acceptable due to the passive nature of the spray nozzles and previous acceptable results. This is a Ginna TS Category (v.b.36) change.
- ix. TS 4.5.2.3.5 - This was revised to only require actuation of the post-accident charcoal filter dampers from an actual or simulated SI signal once every 24 months to ensure that the system aligns itself correctly (SR 3.6.6.15). The post-accident charcoal filter dampers must still be opened at least once per 31 days to allow the system to operate for ≥ 15 minutes. Consequently, only the frequency of the automatic alignment of the dampers is being revised to provide consistency with other specifications. This is a Ginna TS Category (v.b.37) change.
- x. TS 4.5.2.2.a - This was revised to adjust the testing Frequency of the spray additive valves from monthly to once every 24 months consistent with SR 3.6.6.16. This increased testing interval is acceptable since the system only needs to be verified that it can actuate on an actual or simulated SI signal on a refueling basis similar to the SI and RHR systems. Any additional valve testing is addressed by the IST program. In addition, a new Surveillance (SR 3.6.6.12) was added to verify that the CS motor operated isolation valves actuate to their correct position once every 24 months following an actual or simulated SI signal. Finally, a new Surveillance (SR 3.6.6.17) was added to verify that the spray additive flow rate is within limits once every 5 years. These changes ensure that the CS and spray additive tank LCOs continue to be met. These are Ginna TS Category (v.b.38) changes.
- xi. TS 4.5.2.3.3 and 4.5.2.3.4 - These were revised to require that each CRFC unit be operated for ≥ 15 minutes once every 31 days (SR 3.6.6.4). This test will ensure that the CRFC units are OPERABLE in accordance with the LCO. In addition, a new Surveillance is also required once every 24 months to ensure that the CRFC units start on an actual or simulated SI signal. These tests will ensure that the CRFC units are OPERABLE in accordance with the LCO. These are Ginna TS Category (v.a) changes.

xii. TS 4.5.2.3.9 - This was revised to require a test of the automatic actuation capability of the CREATS once every 24 months. This verification is necessary to ensure that the control room environment can be isolated in the event of a radiological release. This is a Ginna TS Category (iv.a) change.

33. Technical Specification 4.6

i. TS 4.6.1.a - The cold or refueling requirements (MODES 5 and 6) for demonstrating DG operability have been revised to include (1) verification of DG day tank fuel oil level, (2) verification of the onsite supply of fuel oil, and (3) operation of the fuel oil transfer system. These are consistent with the required surveillances for DG operability in MODES 1, 2, 3, and 4 and provide assurance that the DG is OPERABLE. This is a Ginna TS Category (iv.a) change.

ii. TS 4.6.1.b.6 - The requirement to verify that the DG is aligned to provide standby power to the associated emergency buses was not added. This requirement is within the definition of an OPERABLE DG and is denoted in the bases of new TS 3.8.1. This is a Ginna TS Category (i) change.

iii. TS 4.6.1.c - The requirement to perform the tests in Specification 4.6.1.b prior to exceeding cold shutdown was not added. This requirement was replaced with a general provision (new SR 3.0.4) that restricts entry into a MODE or other specified condition in the Applicability of an LCO unless the LCO's surveillances have been met. This is a Ginna TS Category (i) change.

iv. TS 4.6.1.d - The diesel fuel oil test requirements were relocated to new TS 5.5.12 and are proposed to be identified as a "program" consistent with the format of NUREG-1431. In addition, the fuel oil testing program was revised to expand the testing requirements consistent with NUREG-1431 and delete the 92 day test of the stored fuel oil. The fuel oil must now be tested before being placed in the storage tanks such that testing of viscosity, water, and sediment after being placed in the storage tanks is no longer required. This is a conservative change which reduces the potential to harm the safety related diesel generators from "bad" fuel oil. This is a Ginna TS Category (v.a) change.

- v. TS 4.6.1.e.1 - The requirement to inspect the DG in accordance with the manufacturer's recommendations was not added. No screening criteria apply for this requirement since DG inspections are not part of the primary success path assumed in the mitigation of a DBA or transient. The requirement does not satisfy the NRC Final Policy Statement technical specification screening criteria and is relocated to the TRM. This is a Ginna TS Category (iii) change.
- vi. TS 4.6.1.e.3(b) - The requirement for DG testing simulating a loss of offsite power in conjunction with a safety injection test signal was revised. Details of the test acceptance criteria were relocated to plant procedures since this level of detail is not typically specified in the SR. This is a Ginna TS Category (iii) change.
- vii. TS 4.6.2.a and 4.6.2.b - The station battery testing requirements were revised to add acceptance criteria, parameters, and associated actions for battery operability supporting DC electrical power subsystems. These requirements are provided in the SRs and enhance the current criteria specified in the TS and is a conservative change regarding the definition of battery OPERABILITY. In addition, the electrolyte temperature is only to be measured every 92 days versus monthly consistent with IEEE-450 requirements. These are Ginna TS Category (iv.a) and (v.a) changes, respectively.
- viii. TS 4.6.2.f. - The details denoting battery degradation were moved to the bases and were revised to include expected life parameters of the battery when compared to a capacity criteria of 100% of the manufacturer's rating. This criteria is used in conjunction with identifying when the surveillance Frequency must be increased and is consistent with ITS. These are Ginna TS Category (iii) and (v.a) changes, respectively.
- ix. TS 4.6.2 - Two new surveillances (SR 3.8.4.1 and SR 3.8.5.1) were added which require verification every 7 days that the battery terminal voltage is ≥ 129 V of float voltage during operating and shutdown conditions. This surveillance ensures that the required battery charger remains capable of maintaining DC system loads and a float charge on the battery. This is a Ginna TS Category (iv.a) change.
- x. TS 4.6.2.c - The requirement for trending battery test data was not added to the new specifications since this is trending must be performed to meet the Frequency requirements for SR 3.8.6.2 and SR 3.8.4.3. Therefore, this requirement is relocated to plant procedures. This is a Ginna TS Category (iii) change.

- xi. TS 4.6.1.b.4 - this was revised to require that the one hour monthly DG run must be performed after successful performance of the monthly DG start (i.e., TS 4.6.1.b.4) or the refueling outage test (i.e., TS 4.6.1.e.4). This ensures that the DG is not being unnecessarily started for the performance of the one hour run. This change is consistent with current testing practices and NUREG-1431. This is a Ginna TS Category (iv.a) change.
- xii. TS 4.6.1.b.4, 4.6.1.e.2, and 4.6.1.e.3 - These were revised to add a note to the surveillance which specifically states that credit may be taken for unplanned events that satisfies these SRs. This is consistent with current operating practice and NUREG-1431 since if a loss of offsite power were to occur requiring a DG run, it should be able to satisfy the surveillance if it meets all of the testing requirements. This also prevents unnecessary tests of the DGs which can lead to potential degradation. This is a Ginna TS Category (v.c) change.

34. Technical Specification 4.7

- i. TS 4.7 was revised to include a surveillance to ensure that each MSIV can close on an actual or simulated actuation signal every 24 months consistent with NUREG-1431 and current Ginna Station TS Table 3.5-2 which require that the isolation signals to the MSIVs be OPERABLE. In addition, Required Actions were provided in the event that the MSIVs cannot close as required by this Surveillance. These actions require restoration of, or closure of an inoperable MSIV, within 24 hours. In the event that both MSIVs are inoperable, the plant must enter LCO 3.0.3. Finally, requirements for the main steam non-return check valves were added. These are Ginna TS Category (iv.a) changes.

35. Technical Specification 4.8

- i. TS 4.8.1 and 4.8.2 - The Frequency of the AFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.

- ii. TS 4.8.3 - This Surveillance was revised to relocate the Frequency of testing the AFW suction and discharge valves to the Inservice Testing Program which provides sufficient control of these testing activities. In addition, the cross-over motor operated isolation valves were not added to the new specifications since these valves are not credited in the accident analyses (see bases for new LCO 3.7.5). These are Ginna TS Category (iii) and (v.b.39) changes, respectively.
- iii. TS 4.8.4 - The Frequency of the SAFW pump tests was changed from monthly to as defined in the Inservice Testing Program consistent with ASME, Section XI requirements. The acceptance criteria was also relocated to Inservice Testing Program consistent with NUREG-1431. This program provides sufficient control for these testing activities. In addition, all OPERABILITY requirements (e.g., required pump flowrates) were relocated to the LCO bases consistent with the ITS Writer's Guide. These are Ginna TS Category (iii) and (i) changes, respectively.
- iv. TS 4.8.5 - This Surveillance was revised to relocate the Frequency of testing the SAFW suction, discharge, and cross-over valves to the Inservice Testing Program which provides sufficient control of these testing activities consistent with NUREG-1431. This is a Ginna TS Category (iii) change.
- v. TS 4.8.6 - This was revised to relocate the acceptance criteria for the AFW and SAFW tests to the actual procedures performing these tests. The new bases identify what is required for OPERABILITY of the AFW and SAFW Systems such that specifying this acceptance criteria is unnecessary. In addition, both the bases and test procedures are controlled under 10 CFR 50.59. This is a Ginna TS Category (iii) change.
- vi. TS 4.8 - A new Surveillance was added requiring verification every 31 days of the correct position of each AFW and SAFW manual, power operated and automatic valve in the flow path that is not locked, sealed or otherwise secured in position. This verification is required to ensure that the AFW and SAFW Systems are OPERABLE when not in service. This is a Ginna TS Category (iv.a) change.
- vii. TS 4.8.10-- The requirement to measure the response time of the AFW pumps and valves to be ≤ 10 minutes once every 18 months was not added to the new specifications. The time requirements for the AFW System are described in the new bases. While some accidents do not require AFW for 10 minutes, the small break LOCA and loss of feedwater transients require AFW within much shorter time frames. Therefore, this Surveillance is not accurate and is not required. This is a Ginna TS Category (v.b.40) change.

36. Technical Specification 4.9

- i. TS 4.9 - This was revised to include an LCO requirement that the measured core reactivity be within 1% $\Delta k/k$ of the predicted values and to add a specific surveillance Frequency of every 31 EFPD after the initial normalization. The Surveillance Requirement was divided into two surveillances to clarify the difference between the initial normalization and the monthly verification. These are Ginna TS Category (v.c) changes.

37. Technical Specification 4.10

- i. TS 4.10.1 and Table 4.10-1 - The requirements for the radiological environmental program which provides measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.
- ii. TS 4.10.2 - The requirements for the land use census which supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iii. TS 4.10.3 - The requirements of the interlaboratory comparison program which confirms the accuracy of the measurements of radiation and of radioactive materials in specified exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public were not added. This program is not related to protection of the public from any DBA or transient analysis. Further, this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

38. Technical Specification 4.11

- i. TS 4.11.1 - The requirements denoting the Frequency and conditions of the SFP filtration system tests were not added. The level of detail is relocated to the VFTP described in new Specification 5.5.10. This is a Ginna TS Category (iii) change.
- ii. TS 4.11.1.1.a, 4.11.1.1.b, and 4.11.1.1.c - These charcoal adsorber system testing requirements were relocated to the VFTP described in the Administrative Controls (TS 5.5.10). This is a Ginna TS Category (i) change.
- iii. TS 4.11.1.1.d - This was not added to the new specifications since this verification is not required to ensure that initial assumptions of the accident analyses are still met. The SFP Charcoal Absorber System does not utilize heaters. The bases for SR 3.7.13.1 state that operating the ventilation system for ≥ 15 minutes every 31 days for systems without heaters is to ensure system operation. In accordance with new LCO 3.7.10 (NUREG-1431 LCO 3.7.13), the ABVS is required to be in operation during fuel movement within the Auxiliary Building. As such, the ABVS is not a standby system at Ginna Station (i.e., the system must be both OPERABLE and in operation during its MODE of Applicability). Therefore, a monthly verification provides no verification of any accident analysis assumption. Instead, two new Surveillances were added which require verification every 24 hours that the Auxiliary Building operating floor level is at a negative pressure with respect to the outside environment and that the ventilation system is in operation. These verifications are consistent with plant practices and ensures that initial assumptions of the fuel handling accident are being maintained. The change is also consistent with Reference 55. This is a Ginna TS Category (v.c) change.

- iv. TS 4.11.2.1 - This was revised to only require verification of RHR pump OPERABILITY once every 12 hours versus 4 hours consistent with SR 3.9.3.1. A Frequency of 12 hours is adequate due to the alarms and indications available to the operators with respect to RHR pump and loop performance. This is a Ginna TS Category (v.b.41) change.
- v. TS 4.11.2.2 - This was revised to remove the requirement for an Inservice Test of the RHR pumps. An Inservice Test should not be required for an operating pump. The status of a non-operating RHR pump is assured by new SR 3.9.4.2 which requires the verification of the breaker alignment and indicated power available to the pump. The Inservice Testing program test is mainly performed to ensure adequate performance during accident conditions which far exceeds the requirements during normal conditions. This test is not necessary to ensure OPERABILITY during MODE 6 operations. This is a Ginna TS Category (v.b.42) change.
- vi. TS 4.11.3.1 - This was revised to only require a verification of the water level in the reactor cavity within 24 hours of fuel movement versus 2 hours. The new TS usage rules state that a SR is to be continuously performed at its required Frequency. However, the SR is only required to be performed when in the MODE of Applicability. Therefore, a SR with a Frequency of 24 hours must have been performed within 24 hours before entering the MODE of Applicability. A Frequency of 24 hours is acceptable due to the large volume of water available and the procedural controls in place. This is a Ginna TS Category (v.c) change.

39. Technical Specification 4.12

- i. TS 4.12.1.1 and Table 4.12-1 - The requirements for radioactive material released in liquid effluents to unrestricted areas which are limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, were not added. No screening criteria apply for these requirements because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- ii. TS 4.12.1.2 - The requirements for the liquid radwaste treatment system which controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, GDC 60 and 10 CFR Part 50, Appendix I, Section II.D, were not added. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iii. TS 4.12.2.1 and Table 4.12-2 - The requirements which assure compliance with 10 CFR Part 20 for the dose rate due to radioactive material released in gaseous effluents beyond the site boundary were not added. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, gaseous effluent dose rate during normal operation is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- iv. TS 4.12.2.2 - The requirements for dose due to noble gases released in gaseous effluents during normal operation over extended periods were not added. These limits are not related to protection of the public from any DBA or transient analysis. Further, gaseous effluents dose (noble gas) values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

- v. TS 4.12.3 - The requirements for the gaseous waste treatment system which reduces the activity level in gaseous waste prior to discharge to the environs were not added. The ventilation exhaust system is not assumed in the analysis of any DBA or transient. Further, the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM and the Radioactive Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. This is a Ginna TS Category (iii) change.

40. Technical Specification 4.13

- i. TS 4.13 - The requirements for periodic testing of leakage for radioactive sources were not added. The source leak test are not assumed in the analysis of any DBA or transient. Further, the leakage from radioactive sources is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function do not satisfy the NRC Final Policy Statement technical specification screening criteria and are relocated to the ODCM. This is a Ginna TS Category (iii) change.

41. Technical Specification 4.14

- i. TS 4.14 - The requirements for the testing of snubbers were not added. Since snubbers testing is controlled within the Inservice Testing Program, the level of detail is relocated to Inservice Testing Program described in new Specification 5.5.7 and more generic program description is provided. This is a Ginna TS Category (iii) change.

42. Technical Specification 4.15

None.

43. Technical Specification 4.16

- i. TS 4.16 - A new surveillance was added which requires verification once within 12 hours and every 12 hours thereafter that an accumulator's motor operated isolation valve is closed when its pressure is greater than or equal to the pressure allowed by the P/T limit curves provided in the PTLR consistent with SR 3.4.12.3. In addition, a verification once within 12 hours and every 31 days thereafter that power is removed to these isolation valves is also added. These verifications are needed to ensure that the accumulator does not discharge into the RCS and cause an overpressure event which challenges the LTOP System. This is a Ginna TS Category (iv.a) change.

- ii. TS 4.16.1.a - This surveillance was revised to delay performance of the PORV functional channel test until 12 hours after decreasing to the LTOP enable temperature specified in the PTLR instead of within 31 days prior to entering the LTOP System Applicability. This change eliminates the performance of the functional test when RCS is between 330°F (the LTOP enable temperature) and 350°F (MODE 3 lower limit) during forced shutdowns. Instead, the test can be performed within 12 hours of entering the specified condition and reduces the immediate operator burden. This is a Ginna TS Category (v.b.43) change.
44. Technical Specification 5.1
- i. TS 5.1.1, TS 5.1.2, and Figure 5.1-1 - The description and figure of the site area boundary was not added to the new specifications consistent with Traveller CEOG-03, C.1. Since the description of this design feature does not meet the criteria for Design Features described in 10 CFR 50.36, this description is relocated to licensee controlled documents (i.e., UFSAR, Section 2.1.2). The figure and description of the exclusion area boundary was also replaced with a table describing this feature consistent with Traveller CEOG-03, C.1. There are Ginna TS Category (iii) changes.
45. Technical Specification 5.2
- i. TS 5.2 - The description of the containment design features was not added. Specific containment features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Therefore, this description is relocated to licensee controlled documents (i.e., UFSAR Sections 3.8.1 and 6.2). This is a Ginna TS Category (iii) change.
46. Technical Specification 5.3
- i. TS 5.3.1.a and TS 5.3.1.c - The description of the reactor core design features was revised consistent with the standard guideline of NUREG-1431. The section now includes the amount, kind, and source of nuclear material related to the reactor core. This is a Ginna TS Category (v.c) change.
 - ii. TS 5.3.1.b - The description of the fuel storage design feature with respect to the maximum enrichment weight percent was revised and relocated to new Specification 4.3.1. The changes are in accordance with the changes discussed in item 47.ii, below. These are Ginna TS Category (v.c) and (i) changes, respectively.

- iii. TS 5.3.2 - The description of the reactor coolant system (RCS) design features was not added. Specific RCS features are covered in the Technical Specification LCO's and, therefore, does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Therefore, this description is relocated to licensee controlled documents (i.e., UFSAR Section 3.7.1 and Chapter 5). This is a Ginna TS Category (iii) change.
- iv. TS 5.3.1.b - This was revised to increase the fuel enrichment limit from 4.25 weight percent to 5.05 weight percent. This change has been evaluated and found to be acceptable with respect to postulated fuel handling accidents (Ref. 29). This is a Ginna TS Category (v.b.46) change.

47. Technical Specification 5.4

- i. TS 5.4.1, 5.4.2, 5.4.6, and Figures 5.4-1 and 5.4-2 - The description of the fuel storage design features denoting spent fuel storage regions and borated water concentrations were relocated to Chapters 3.7 and 3.9. These features are discussed in LCOs 3.7.11, 3.7.12, 3.7.13, and 3.9.1 as appropriate. In addition, appropriate Required Actions were added in the event that SFP water level, boron concentration, or SFP region storage requirements are not met. This is a Ginna TS Category (i) change.
- ii. TS 5.4.2 - The description of the fuel storage design features was revised. The revision to these features are based on a revised criticality analysis supporting the proposed 18 month fuel cycle (Reference 29). The description of these features follow the standard guideline of NUREG-1431 which would include the amount, kind, and source of special nuclear material with the exception that nominal center to center spacing between the fuel assemblies was not added. This is a Ginna TS Category (v.c) change.

- iii. TS 5.4.3 - The description of the fuel storage design feature denoting the 60-day limit on storage of discharged fuel assemblies in Region 2 was not added. No screening criteria applies for the time limit on storage of discharged fuel assemblies in Region 2. The current 60-day limit was established to provide sufficient margin in spent fuel pool temperature calculations as a result of decay heat loads in Region 2 from discharged fuel assemblies (Reference 39). Although the spent fuel pool cooling system and, thus, the associated restriction on heat load prevent structural integrity damage to the spent fuel pool, they are not assumed to function to mitigate the consequences of a design basis accident (DBA). The restriction on heat load is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. The restriction on heat load is a non-significant risk contributor to core damage frequency and offsite doses. Since this does not meet the criteria for Design Features described in 10 CFR 50.36(c)(4) and no NRC Final Policy Statement technical specification screening criteria apply, this requirement is relocated to the TRM. This is a Ginna TS Category (iii) change.
- iv. TS 5.4.4 and 5.4.5 - These were revised consistent with References 29 and 39 to provide the amount, kind, and source of material which is stored in the canisters. This is a Ginna TS Category (v.c) change.
- v. TS 5.4 - This was revised to include descriptions of the SFP drainage system and capacity. This information is currently contained in the bases for this section. Since NUREG-1431, Chapter 4 does not contain any bases, this information has been relocated to the specification. This is a Ginna TS Category (i) change.

48. Technical Specification 5.5

- i. TS 5.5 - The description of the waste treatment systems design features was not added. No screening criteria apply for the description of these features. Specific waste treatment systems features are either covered in the Technical Specification LCO's or have been relocated to other licensee controlled documents and, therefore, do not meet the criteria for Design Features described in 10 CFR 50.36(c)(4). Since the description of these design features does not satisfy the NRC Final Policy Statement technical specification screening criteria, this description is relocated to licensee controlled documents (i.e., UFSAR Chapter 11). This is a Ginna TS Category (iii) change.

49. Technical Specification 6.1

- i. TS 6.1.1 - The requirement was revised to include a statement that the plant manager shall approve each proposed test, experiment or modification to structures, systems or components that affect nuclear safety. This is a Ginna TS Category (iv.a) change.
- ii. TS 6.1 - A new requirement (Specification 5.1.2) was added which establishes the requirement for Shift Supervisor responsibility. This is a Ginna TS Category (iv.a) change.
- iii. The plant manager title was revised to be more generic consistent with Reference 62. See also item 50.ii below. This is a Ginna TS Category (vi) change.

50. Technical Specification 6.2

- i. Cross references to existing regulatory requirements are redundant and generally not incorporated into NUREG-1431. This is a Ginna TS Category (ii) change.
- ii. Plant specific management position titles in the current Technical Specifications are replaced with generic titles consistent with Reference 62. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed TS 5.3, and compliance details relating to the plant specific management position titles are identified in the UFSAR. The two major specific replacements are the generic "plant manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "a corporate vice president" in place of the specific Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in the UFSAR which has specific regulatory review requirements for changes. This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change.
- iii. TS 6.2.1.d - The requirement describing the capability of training, health physics and quality assurance to have direct access to responsible corporate management was modified to be consistent with NUREG-1431. These modifications are editorial changes only which do not change the intent or requirements of this specification. This is a Ginna TS Category (vi) change.

- iv. TS 6.2.2.b - The requirements describing the required operating crew compositions were not added. These requirements are specified in 10 CFR 50.54(k), (l), and (m) and proposed TS 5.2.2.a, 5.2.2.b, and 5.2.2.e. This is a Ginna TS Category (ii) change.
- v. TS 6.2.2.d - The requirement was revised to clarify that the individual qualified in radiation protection procedures is allowed to be absent for not more than two hours. This is consistent with the requirements for shift crew composition. This is a Ginna TS Category (v.c) change.
- vi. TS 6.2.2.e - The requirement describing the overtime requirement for plant staff who perform safety related functions was revised to reference a NRC approved program for controlling overtime. This is a Ginna TS Category (vi) change.
- vii. A new requirement was added which specifies that the Operations Manager or Operations middle manager shall hold a SRO. This change is consistent with NUREG-1431 and ensures that at least one operations manager holds a SRO. This is a Ginna TS Category (iv.a) change.

51. Technical Specification 6.3

- i. TS 6.3.1 - The reference to the RG&E letter dated December 30, 1980, was replaced with wording considered more appropriate. The current STA program at Ginna Station is discussed in References 40 and 42 and was reviewed and approved by the NRC. The revised wording eliminates the need to revise the Technical Specifications if the STA program is later revised, but still requires NRC approval of these changes. This is a Ginna TS Category (vi) change.

52. Technical Specification 6.4

- i. TS 6.4 - The requirements for a Training Program were not added. The requirements are either adequately addressed by other Section 5.0 administrative controls or are addressed by 10 CFR 55 requirements. Therefore, these requirements are relocated to the UFSAR. This is a Ginna TS Category (iii) change.

53. Technical Specification 6.5

None.

54. Technical Specification 6.6

None.

55. Technical Specification 6.7

- i. TS 6.7.1.a - The initial operator actions for Safety Limit (SL) violations were revised as follows:
 - a. For violation of the Reactor Core or RCS Pressure SL in MODES 1 and 2, the requirement to immediately shutdown the reactor (effectively to be in MODE 3) was revised to allow 1 hour to restore compliance and place the unit in MODE 3. Immediately shutting down the reactor could infer action to immediately trip the reactor. The revision provides the necessary time to shutdown the unit in a more controlled and orderly manner than immediately tripping the reactor which could result in a plant transient. The proposed time continues to minimize the time allowed to operate in MODE 1 or 2 with a SL not met. This is a Ginna TS Category (v.b.44) change.
 - b. For violation of the RCS Pressure SL in MODES 3, 4, and 5, an additional action was added which requires restoring compliance with the SL within 5 minutes. Specifying a time limit for operators to restore compliance provides greater guidance to plant staff. This is a Ginna TS Category (v.a) change.
- ii. TS 6.7.1.b - The requirement for notification to management personnel and the offsite review function of a SL violation was not added to the new specifications. Notification requirements are relocated to the TRM. This is a Ginna TS Category (iii) change. The requirement for notification to the NRC of a SL violation was not added to the new specifications since this requirement is denoted in 10 CFR 50.36 and 10 CFR 50.72. This is a Ginna TS Category (ii) change.
- iii. TS 6.7.1.c - The requirement that a Safety Limit Violation Report be prepared was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the onsite review committee to review the Safety Limit Violation Report was not added to the new specifications. The responsibilities of the onsite review committee are relocated to the TRM. This is a Ginna TS Category (iii) change. SL violations are reported to the NRC in accordance with the provisions of 10 CFR 50.73. The details describing the requirements for content of the Safety Limit Violation Report is, therefore, controlled by the provisions of 10 CFR 50.73 and does not need to be specified in TS. This is a Ginna TS Category (ii) change.

- iv. TS 6.7.1.d - The requirement for the submittal of a Safety Limit Violation Report to the NRC was not added to the new specifications. This is a duplication of requirements denoted in 10 CFR 50.36 and 10 CFR 50.73. This is a Ginna TS Category (ii) change. The requirement for the submittal of a Safety Limit Violation Report to management personnel and the offsite review function was not added to the new specifications. The distribution of reports submitted in accordance with 10 CFR 50.73 are relocated to the TRM. This is a Ginna TS Category (iii) change.

56. Technical Specification 6.8

- i. TS 6.8.1.d - The Offsite Dose Calculation Manual implementation is covered by a more generic item which is specified in Section 5.5. It is not necessary to specifically identify each program under procedures (see Section D, item 56.iv). Since the requirements remain, this is considered to be a change in the method of presentation only. This is a Ginna TS Category (i) change.
- ii. TS 6.8.1.e - The PCP description was not added since this program only implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71 and does not impose any new regulations. The detailed description of the PCP is provided in licensee controlled documents. This is a Ginna TS Category (iii) change.
- iii. TS 6.8.1 - A new specification (TS 5.4.1.b) was added which establishes the requirement for written emergency operating procedures implementing the requirements of NUREG-0737 and NUREG-0737, Supplement 1. This is a Ginna TS Category (iv.a) change.

- iv. TS 6.8.1 - A new specification (TS 5.4.1.e) was added which establishes the requirement for written procedures for programs and manuals denoted in new Specification 5.5. These Programs include:

<u>ITS</u>	<u>Current TS</u>	<u>Program</u>
5.5.1	1.13 & 6.15	Offsite Dose Calculation Manual
5.5.2	4.4.3	Primary Coolant Sources Outside Containment
5.5.3	New	Post Accident Sampling Program
5.5.4	3.9 & 3.16	Radioactive Effluent Controls Program
5.5.5	New	Component Cyclic or Transient Limit
5.5.6	4.4.4	Pre-Stressed Concrete Containment Tendon Surveillance Program
5.5.7	4.2	Inservice Testing Program
5.5.8	4.2	Steam Generator (SG) Tube Surveillance Program
5.5.10	4.5.2.3 & 4.11.1	Ventilation Filter Testing Program
5.5.11	3.9.2.5 & 3.9.2.6	Explosive Gas and Storage Tank Radioactive Monitoring Program
5.5.12	4.6.1.d	Diesel Fuel Oil Testing Program
5.5.13	New	Technical Specification Bases Control
5.5.14	New	Safety Function Determination Program
5.5.15	New	Containment Leakage Rate Testing Program

The technical content of several requirements are being moved from other chapters of the current Technical Specifications and are proposed to be identified as Programs in accordance with the format of NUREG-1431. This is a Ginna TS Category (i) change. Other programs were added, except as discussed below, to ensure consistency in the implementation of required programs within the current licensing basis. The Radioactive Effluent Controls Program was added due to the relocation of the radiological Technical Specifications consistent with Generic Letter 89-01 and the changes to 10 CFR 20. The Bases Control program was added to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program was added to support implementation of the support system operability characteristics of the Technical Specifications (new LCO 3.0.6). These are Ginna TS Category (iv.a) changes:

- v. TS 6.8.1.c - The radiological environmental monitoring program is covered by a more generic item which is specified in specification 5.5. It is not necessary to specifically identify each program under procedures (see Section D, item 56.iv). Since the requirements remain, this is considered to be a change in the method of presentation only. This is a Ginna TS category (i) change.

57. Technical Specification 6.9

- i. TS 6.9 - The reference to reporting requirements were revised consistent with 10 CFR 50.4. This is a Ginna TS Category (vi) change.
- ii. TS 6.9.1.1 - The requirement to submit a Startup Report was not added. The Startup Report is more appropriately addressed in the NRC Safety Evaluation Report authorizing an Operating License, increased power level, installation of a new nuclear fuel design or manufacturer, or modifications which significantly alter the nuclear, thermal, or hydraulic performances of the plant. The Startup Report is required to be submitted within 90 days following completion of the above activities and does not require NRC approval. Therefore, inclusion of the requirement for this report in Technical Specifications is not necessary to assure safe plant operation. This is a Ginna TS Category (ii) change.
- iii. TS 6.9.1.2 - The requirements describing the details of the monthly report were not added. These details are appropriately relocated to procedures or other licensee controlled documents. This is a Ginna TS Category (iii) change.
- iv. TS 6.9.1.3, TS 6.9.1.4, Table 6.9-1 and Table 6.9-2 - The details and methods implementing these specifications were not added. These details are appropriately relocated to the ODCM and the Effluent Controls Program described in new Specifications 5.5.1 and 5.5.4, respectively. The submittal date was also changed to May 15th to allow the submittal of the Annual Radiological Environmental Operating Report to correspond with the Monthly Operating Report submittal date. This is a Ginna TS Category (iii) change.
- v. TS 6.9.1.4 - The specific date referenced for the annual submittal was revised consistent with the requirements of 10 CFR 50.36a. This is a Ginna TS Category (vi) change.
- vi. TS 6.9.1.5 - The requirement for the reporting of challenges to pressurizer PORVs or safety valves was revised from an annual to a monthly report and relocated to the Monthly Operating Report (new Specification 5.6.4). This is a Ginna TS Category (v.c) change.
- vii. TS 6.9.2.1 - The reporting requirement related to sealed sources was not added since this is specified in 10 CFR 30.50. The detailed description of these reporting requirements are provided in licensee controlled documents. This is a Ginna TS Category (iii) change.

- viii. TS 6.9.2.4 - The reporting requirement for reactor overpressure protection system operation was revised. The reporting requirement is detailed in proposed Specification 5.6.4, and is generally included in the LER requirements to report a RCS pressure transient that exceeds expected values or that is caused by unexpected factors. Since the criteria identified in 10 CFR 50.73 includes the area of degraded boundaries that necessitates reporting, any minor differences are negligible with regard to safety. This is a Ginna TS Category (ii) change.
- ix. A new requirement TS 5.6.5 was added which establishes the reporting requirement for the COLR. The COLR is required due to the removal of existing Technical Specification core operating limits. This is a Ginna TS Category (iv.a) change.
- x. A new requirement TS 5.6.6 was added which establishes the reporting requirement for the RCS PTLR. The PTLR is required due to the removal of existing Technical Specification pressure and temperature operating limits. This is a Ginna TS Category (iv.a) change.

58. Technical Specification 6.10

None.

59. Technical Specification 6.11

None.

60. Technical Specification 6.12

None.

61. Technical Specification 6.13

- i. TS 6.13.1 - Plant-specific position titles in the current Ginna Station TS were replaced with generic titles (i.e., radiation protection technician). The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in the UFSAR which has specific regulatory review requirements for changes. This change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions. This is a Ginna TS Category (vi) change.

62. Technical Specification 6.14

None.

63. Technical Specification 6.15

- i. TS 6.15.1.b - The approval process for ODCM changes was revised to clarify that the effective changes be approved by the plant manager instead of the onsite review function. Since the onsite review function reports to the Plant Manager, this is a conservative change. This is a Ginna TS Category (v.a) change.

64. Technical Specification 6.16

- i. TS 6.16 - The process for changes to the PCP was not added to the new specifications since this program only implements the requirements of 10 CFR Part 20, 10 CFR Part 61, and 10 CFR Part 71 and does not impose any new requirements. The detailed description of the PCP is provided in licensee controlled documents. This is a Ginna TS Category (iii) change.

65. Technical Specification 6.17

- i. TS 6.17 - The requirements for major changes to radioactive waste treatment systems was not added. Changes to these systems are controlled by 10 CFR 50.59. NRC notification of significant changes to these systems is addressed by 10 CFR 50.59(b)(2). Therefore, this specification is relocated to the TRM. This is a Ginna TS Category (iii) change.

66. New Requirements (Ginna TS Category (iv.a) Changes)

- i. LCO 3.4.1 and the associated surveillance requirements were added for DNB limits. This new requirement places limits on pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure that the minimum DNBR will be met for all analyzed transients.
- ii. LCO 3.7.3 and the associated surveillances were added for the MFW pump discharge valves (MFPDVs), MFW regulating valves, and the associated bypass valves. This new requirement specifies an isolation time of 80 seconds for the MFPDVs and 10 seconds for the remaining valves and requires them to be OPERABLE above MODE 4 to provide isolation capability as assumed in the accident analyses.
- iii. LCO 3.7.4 and the associated surveillance were added for the atmospheric relief valves (ARVs). The LCO requires that the ARVs be OPERABLE when RCS average temperature is > 500°F in MODE 3 to provide cooldown capability following a SGTR event as assumed in the accident analyses. A Surveillance to verify that each ARV is capable of opening and closing once every 24 months was also added.

iv. A COLR was developed which contains the actual limits for LCOs associated with reactor physic parameters that may change with each refueling. To prevent the need to revise Technical Specifications for parameters which are calculated using NRC approved methodology, Generic Letter 88-16 (Ref. 56) allows these limits to be relocated from the technical specifications. A copy of the proposed Ginna Station COLR is provided in Attachment F. The following parameters were relocated to the COLR:

- a. SHUTDOWN MARGIN
- b. MODERATOR TEMPERATURE COEFFICIENT
- c. Shutdown Bank Insertion Limit
- d. Control Bank Insertion Limits
- e. Heat Flux Hot Channel Factor
- f. Nuclear Enthalpy Rise Hot Channel Factor
- g. AXIAL FLUX DIFFERENCE
- h. Not used
- i. RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits
- j. Not used
- k. Not used
- l. Not used
- m. Refueling Boron Concentration

v. A RCS PTLR was developed which contains the actual limits for LCOs associated RCS pressure and temperature limits and LTOP. To prevent the need to revise Technical Specifications for parameters which are calculated using NRC approved methodology, NUREG-1431 allows these limits to be relocated from the technical specifications. A copy of the proposed Ginna Station PTLR is provided in Attachment G. The following parameters were relocated to the PTLR:

- a. RCS Pressure and Temperature Limits
- b. Low Temperature Overpressure Protection (LTOP) System Enable Temperature
- c. LTOP Setpoint

67. License

- i. The license was revised to relocate requirements associated with Secondary Water Chemistry Monitoring Program, Systems Integrity, and Iodine Monitoring to Appendix A of the license (i.e., TS). Changes to both the license and TS require NRC approval such that there is no reduction in commitment with respect to this change. This is a Ginna TS Category (i) change.
- ii. Minor editorial changes were made to provide consistency within the license. These are administrative changes only which do not change the intent of the license. These are Ginna TS Category (vi) changes.



- iii. The exemption to 10 CFR 50.48(c)(4) was removed from the license since this exemption expired in 1986 and is no longer required. This is a Ginna TS Category (vi) change.
- iv. The exemption to 10 CFR 50.46(a)(1) was removed from the license since this exemption is no longer required since the ECCS models for Ginna Station have since been revised. This is a Ginna TS Category (vi) change.

E. SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D are organized into 6 categories and subcategories as necessary. These categories of changes are evaluated with respect to 10 CFR 50.92(c) and shown to not involve a significant hazards consideration as described below.

E.1 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - ADMINISTRATIVE CHANGES

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (i), (ii), (v.c), or (vi) changes do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes involve either (1) the relocation of requirements within the Technical Specifications to support consolidation of similar requirements, (2) the reformatting, renumbering or rewording of the existing Technical Specifications to provide consistency with NUREG-1431, (3) the deletion of duplicate regulatory requirements, or (4) minor changes to the Technical Specifications such that the changes do not involve any technical issues. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. These changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.2 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - RELOCATED SPECIFICATIONS

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iii) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document (e.g., Technical Requirements Manual or UFSAR) which will continue to be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions in the Administrative Controls Section of the Technical Specifications. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.
2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of safety because the changes do not impact any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, the majority of changes are consistent with the Westinghouse Standard Technical Specification, NUREG-1431, which has been approved by the NRC Staff. Therefore, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety. For those requirements proposed to be relocated which are retained within NUREG-1431, the relocated items are similar in nature to other relocated requirements or are not credited in the accident analyses for Ginna Station.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.3 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - MORE RESTRICTIVE CHANGES

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.a) and (v.a) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes provide more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes do impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change in Section D, each change in this category is by definition providing additional restrictions to enhance plant safety. The change maintains requirements within safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

E.4 SIGNIFICANT HAZARDS CONSIDERATION EVALUATION - LESS RESTRICTIVE CHANGES

LESS RESTRICTIVE CHANGE CATEGORY (iv.b.1)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.b.1) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Required Actions of the Diesel Generator (DG) Loss of Power (LOP) start instrumentation (current Table 3.5-1, Functional Units # 18 and #19) from an action to shutdown to an action to restore the channel to an OPERABLE status or enter the applicable conditions for an inoperable DG. The start instrumentation function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the OPERABLE start instrumentation channels from performing their intended function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that are no more restrictive than actions for the loss of one DG. The change maintains requirements within safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (iv.b.2)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (iv.b.2) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the actions for an inoperable DG to: (1) eliminate the testing of the OPERABLE DG if, within 24 hours, it can be determined that the OPERABLE DG is not inoperable due to a common cause failure, and (2) eliminate the requirement to test the OPERABLE DG once every 24 hours until the second DG is restored to OPERABLE status (TS 3.7.2.2.b.1). The testing requirements for an OPERABLE DG are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not degrade the capability of the OPERABLE DG from performing its intended function since some DG failures can be conclusively determined not to apply to a second DG without requiring excessive testing. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that prevent unnecessary DG starts which can potentially adversely affect DG reliability. The change maintains DG OPERABILITY requirements within the safety analyses and licensing bases. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.1)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.1) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Refueling Frequency which is used to define CHANNEL CALIBRATION and other testing intervals, from 18 months to 24 months (TS 1.12 and 4.4.6.2). The Frequency between CHANNEL CALIBRATIONS is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. TS required equipment is current maintained under a Reliability Centered Maintenance program such that their failures are tracked and trended. In addition, instrumentation setpoints and equipment history have been verified to be acceptable with respect to this change. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The equipment testing intervals are increased, but they still must be maintained OPERABLE consistent with their TS requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.2)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.2) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the applicability associated with the RCS Safety Limits (SL) in MODE 6 (current TS 2.2). Adequate margin exists such that it is not possible to pressurize the RCS greater than the SL pressure while in MODE 6. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these limits are not credited for mitigation of any accident in the omitted condition. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since it is not possible to pressurize the RCS greater than the SL pressure while in MODE 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.3)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.3) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement for the pressurizer water level lower limit of 12% (current TS 3.1.1.5.a). This requirement relates to a reactor trip function that was removed at Ginna Station as a result of IE Bulletin 79-06A (Ref. 45). Therefore, this change does not significantly increase the probability of a previously analyzed accident nor significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant since the trip function has already been removed. The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
- 3.. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the pressurizer low level trip function is no longer credited. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.4)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.4) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the applicability and deletes requirements associated with the overpressurization protection function of the pressurizer safety valves in MODES 5 and 6 (current TS 3.1.1.3.a and TS 3.1.1.3.b). The pressurizer safety valves do not perform a safety function in the omitted conditions. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these limits are not credited for mitigation of any accident in the omitted conditions. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since these valves do not perform a safety function in MODES 5 and 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.5)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.5) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change deletes the requirements associated with SG temperature and pressure variables (current TS 3.1.1.2 and TS 3.1.2.2). The temperature and pressure variables are not specifically modeled in the safety analysis except through the variables of RCS pressure, temperature, and flow which are addressed in the heatup and cooldown rates in the PTLR. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, these SG variables are not credited for mitigation of any accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since all necessary heatup and cooldown rates are addressed by the PTLR. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.6)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.6) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the period of time (from 6 hours to 72 hours) continued operation is allowed prior to confirming through the performance of an engineering evaluation, the structural integrity of the RCS after exceeding pressure or temperature limits (current TS 3.1.2.1.c.1). The requirement is associated with a function that is not an assumed initiator for any accidents previously evaluated since the exceeded limits are subsequently restored. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this function is not credited for mitigation of any accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The change maintains requirements within current safety analyses since the time that out-of-condition limits are restore is not changed. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.7)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.7) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change provides a Note allowing the plant to change MODES if either the containment sump monitor or both the containment atmospheric radioactivity monitors are inoperable (current TS 3.1.5.1). The RCS LEAKAGE detection system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the system to perform its required function since some form of LEAKAGE detection must always remain OPERABLE under these circumstances or a plant shutdown commenced. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since some form of RCS LEAKAGE detection must remain OPERABLE in MODES 1, 2, 3, and 4. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.8)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.8) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows an additional 4 hours to correct administrative and other similar discrepancies in the SG Tube Surveillance Program before commencing a reactor shutdown (current TS 3.1.5.2.2). Administrative discrepancies in the SG Tube Surveillance Program are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the SG tubes to perform their intended function since the limit on SG tube leakage remains. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change results in actions that allow restoration of minor administrative discrepancies without affecting any safety analysis assumptions with respect to SG tube leakage. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.9)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.9) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 72 hours to restore accumulator boron concentration to within acceptable limits versus 1 hour (current TS 3.3.1.1.b and 3.3.1.3). The accumulator boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the accumulator to perform its required function under these circumstances since it will only allow additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The accumulator boron concentration is not as critical feature as other accumulator parameters (e.g., water volume) such that additional time for restoration does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.10)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.10) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 72 hours to restore accumulator boron concentration to within acceptable limits versus 1 hour (current TS 3.3.1.1.a and 3.3.1.2). The RWST boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the RWST to perform its required function under these circumstances since it will only allow additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The RWST boron concentration is not as critical feature as other RWST parameters (e.g., water volume) such that additional time for restoration does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.11)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.11) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change: (1) allows both SI pump flow paths to be isolated for up to 2 hours to perform pressure isolation valve testing, and (2) allows up to 4 hours, or until the RCS cold legs exceed 375°F, to place into service ECCS pumps declared inoperable due to LTOP considerations (current TS 3.3.1.1.c). The ECCS System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change allows required testing to be performed on the ECCS and reduces the potential for a transient to challenge the LTOP System. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change allows required testing to be performed on the ECCS, reduces the potential for a transient to challenge the LTOP Systems, and are consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.12)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.12) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change provides an AOT of 72 hours for two inoperable post-accident charcoal filter trains (current TS 3.3.2.2). The system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the system to perform its required function under these circumstances. This will allow an additional time to restore the system to an OPERABLE status prior to initiating a plant shutdown. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the CRFC units which supply the post-accident charcoal filter trains may be removed from service for up to 7 days prior to initiating a plant shutdown. In addition, the 100% redundant CS trains must remain OPERABLE in this condition. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.13)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.13) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the CCW heat exchanger requirements to allow 1 heat exchanger to be inoperable for up to 31 days versus 24 hours (current TS 3.3.3.1). The CCW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the CCW system to perform its required function under these circumstances since the heat exchanger is a passive device similar to the CCW piping. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the CCW piping is also a passive device, which if it were to fail, would result in the loss of the entire CCW System which has been analyzed with acceptable results. Therefore, this change does not involve a significant reduction in a margin of safety.



Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.14)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.14) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the AOT for two motor driven AFW pumps, from 24 hours to 72 hours, to be consistent with that for the turbine driven AFW pump (current TS 3.4.2.1.b). The AFW system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. This change does not further degrade the capability of the AFW system to perform its required function under these circumstances since the turbine driven AFW pump is fully capable of supplying both SGs. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change maintains requirements within current safety analyses since the turbine driven AFW pump is fully capable of supplying both SGs. In addition, for accident conditions in which AFW is not immediately required (i.e., not required for 10 minutes), the SAFW System is available. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.15)

Not used (see Reference 30).

LESS RESTRICTIVE CHANGE CATEGORY (v.b.16)

Not used (see Reference 3.0).

LESS RESTRICTIVE CHANGE CATEGORY (v.b.17)

Not used (see Reference 3.0).

LESS RESTRICTIVE CHANGE CATEGORY (v.b.18)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.18) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the required channels for Diesel Generator (DG) Loss of Power (LOP) start instrumentation (current Table 3.5-1, Functional Units # 18 and #19) from individually specifying the loss of voltage and degraded voltage channels to requiring two channels of undervoltage per 480 V safeguards bus. The start instrumentation function is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change does not further degrade the capability of the OPERABLE DG LOP instrumentation channels from performing their intended function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change only clarifies the actual design of the DG LOP instrumentation without affecting the safety function of the specified channels. The requirement for a loss of voltage and degraded voltage function is specified in the surveillance requirement for this LCO. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.19)

Not used (see Reference 30).

LESS RESTRICTIVE CHANGE CATEGORY (v.b.20)

Not used (see Reference 30).

LESS RESTRICTIVE CHANGE CATEGORY (v.b.21)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.21) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the current AOT to restore inoperable Post Accident Monitors (PAMs), revises the actions for inoperable PAMs that are not restored to service within the AOT, and revises the PAM testing frequencies (current TS 3.5.3, 3.6.4.2, and 4.4.7). The PAMs are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The PAMs are not required to provide automatic response to any design basis accident. The additional time and surveillance frequencies has been evaluated and determined by the NRC to not significantly affect the contribution of the monitors to risk reduction. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.22)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.22) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows the use of a closed system to be used to isolate a penetration with a failed containment isolation valve for up to 72 hours (current TS 3.6.3). The containment isolation system is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the containment isolation system to perform its required function under these circumstances since the closed system is a passive device which is missile protected. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The containment isolation system remains capable of performing its intended function since the closed system is missile protected, leak tested, and capable of maintaining containment integrity in the event of an accident. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.23)

Not used.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.24)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.24) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the AOT for an inoperable 480 V safeguards bus from 1 hour to 8 hours before requiring a plant shutdown (current TS 3.7.2.2.c). The 480 V safeguards buses are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the 480 V safeguards buses to perform their required function under these circumstances since a redundant train is available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Allowing additional time to restore an inoperable 480 V safeguards bus does not adversely affect the accident analyses since a redundant train is available. The increased time is also consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.25)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.25) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to verify power distribution after each refueling from prior to reaching 50% RTP to < 75% RTP (current TS 3.10.2.1). Peaking factors are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Allowing power ascension to 75% RTP before verifying power distribution still provides the necessary margin to ensure design limits are met since peaking factors are most decreased near 100% RTP. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.26)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.26) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the requirement to maintain F_Q and F_{AH} within limits at all times to only in MODE 1 (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. These power distribution limits are not necessary to be met during MODE 2 since there is insufficient energy in the fuel to require these limits. In MODES 3, 4, 5, and 6, the reactor is not critical and, as such, these limits are not required. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.27)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.27) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Completion Time, from 24 hours to 72 hours, to reduce the Overpower ΔT , Overtemperature ΔT , and Power Range Neutron Flux - High trip setpoints when F_Q or F_{AH} is not within limits (current TS 3.10.2.2). These power distribution limits are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function under these circumstances since the Required Actions for these power distribution limits already require a power reduction in direct relationship to the percentage that the limit was exceeded. The reduction of trip setpoints only provides additional protection. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Allowing additional time to reduce the setpoints for associated reactor trip functions only provides secondary protection with respect to potential unanalyzed power distributions since reactor power has already been reduced. Therefore, this change does not involve a significant reduction in a margin of safety. The change for Overpower ΔT is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.28)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.28) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change deletes the requirement to identify the cause of QPTR exceeding 1.02 or limit power to < 50% RTP (current TS 3.10.2.4). The QPTR is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. If the QPTR is not within limits, thermal power is required to be reduced proportional to the percentage that QPTR is outside the limits to compensate for the tilt and flux mapping must be initiated. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Removing the requirement to identify the cause of the tilt or reduce power to < 50% RTP does not adversely affect the accident analyses since a power reduction proportional to the percentage that QPTR is outside the limit is required. It is not always possible to identify the cause of the tilt and the remaining Required Actions already underway are adequate to assure safe operation of the plant. This power change is consistent with NUREG-1431 and WCAP-12159 (Ref. 51). Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.29)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.29) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performance of control rod exercises from monthly to every 92 days (current Table 4.1-2, Functional Unit #6a). Control Rods are only considered as an initiator for rod ejection accidents which are not related to this Surveillance. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function since this Surveillance only confirms normal operational indications of control rod OPERABILITY. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. Control Rod OPERABILITY is normally verified by normal operational practices such that increasing the allowed Surveillance interval does not involve a significant reduction in a margin of safety. The change is also consistent with NUREG-1431 and NUREG-1366 (Ref. 8).

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.30)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.30) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying the NaOH concentration in the spray additive tank from monthly to once every 184 days (current Table 4.1-2, Functional Unit #13). The spray additive tank is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function since the tank is passive with available level indications to the operators which would indicate a change in concentration. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the spray additive tank from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.31)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.31) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing an RCS water inventory balance from daily to once every 72 hours (current Table 4.1-2, Functional Unit #15). Verifying RCS water inventory is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of operations to identify LEAKAGE in the RCS since other indications, including letdown, are available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not degrade the capability of operations to identify LEAKAGE in the RCS since other indications are available. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.32)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.32) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing verification of the SFP boron concentration from once every 31 days to once every 31 days if a verification of fuel storage has not been complete (current Table 4.1-2, Functional Unit #17). Verifying SFP boron concentration is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not affect the accident analyses since boron concentration is only credited during a fuel handling accident prior to the time which the fuel has been verified to be correctly stored. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not affect the assumptions used for a fuel handling accident. Therefore, this change does not involve a significant reduction in a margin of safety. This change (with the exception of the 31 day Frequency) is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.33)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.33) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying the DG fuel oil inventory from daily to once every 31 days (current Table 4.1-2, Functional Unit #16). The DG fuel oil tank is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the system to perform its required function since the tank is passive with available level indications to the operators which would indicate a change in inventory. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the DG fuel oil tank from performing its intended safety function since other indicators are available to operators. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.34)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.34) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying RCS gross specific activity from once every 72 hours to once every 7 days (current Table 4.1-4, Functional Unit #1). Verifying RCS gross specific activity is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of operations to identify fuel failures since other indications, including radiation alarms, are available. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not degrade the capability of operations to identify gross fuel failure since other indications are available. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.35)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.35) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to test the SI cold leg injection and RHR RCS PIVs each cold shutdown greater than 7 days (current TS 4.3.3.1). These valves are normally maintained closed (i.e., they are not relied upon or used during power operation or cold shutdown conditions). Performing testing on these PIVs should only be required once every 24 months or within 24 hours of their being opened since more frequent testing would not likely provide any additional information. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the PIVs to perform their required function since the valves are maintained closed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the RCS PIVS from performing their intended safety function since they will be tested a minimum of once every 24 months. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.36)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.36) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing the spray nozzle gas test from once every 5 years to once every 10 years (current TS 4.5.1.2.b). The spray ring nozzles are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the CS System to perform its required function since the nozzles are passive and located in a generally unaccessible area. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the CS System from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. The revised Frequency is also consistent with NUREG-1431 and NUREG-1366 (Ref. 8).

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.37)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.37) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing actuation testing of the post-accident charcoal filter dampers from monthly to once every 24 months (current TS 4.5.2.3.5). The post-accident charcoal filters are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the post-accident charcoal filters to perform their required function since the dampers have demonstrated a high degree of reliability and the CS System provides a 100% redundant iodine removal capability. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the post-accident filters from performing their intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.38)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.38) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for performing the spray additive valves from monthly once every 24 months (current TS 4.5.2.2.a). The spray additive valves are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the spray additive system from performing its required function since have demonstrated a high degree of reliability and the post-accident charcoal filters provide 100% redundant iodine removal capability. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the spray additive system from performing its intended safety function. The revised Frequency is also consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. The revised Frequency is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.39)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.39) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to test the AFW motor driven pump cross-over motor operated isolation valves (current TS 4.8.3). The AFW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the AFW System since the cross-over isolation valves are not credited in the accident analysis. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The deletion of the AFW cross-over isolation valves testing requirements does not prevent the AFW System from performing its intended safety function since the valves are not credited in the accident analysis. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.40)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.40) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change eliminates the need to perform a verification that the AFW pumps can start within 10 minutes once every 18 months (current TS 4.8.10). The AFW System is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the AFW System from performing its required function since this verification is not consistent with the accident analysis times. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The deletion of this Surveillance does not prevent the AFW System from performing its intended safety function since the 10 minute verification is not consistent with the accident analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.41)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.41) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the Frequency for verifying a RHR pump, is providing forced flow in MODE 6 from once every 4 hours to once every 12 hours (current TS 4.11.2.1). Verification of RHR pump OPERABILITY is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the RHR System to provide decay heat removal since there are numerous indications available to plant operators of a loss of an RHR pump. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change in Surveillance Frequency does not prevent the RHR System from performing its intended safety function. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.42)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.42) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to perform Inservice Testing surveillances of the RHR pumps during MODES 5 and 6 (current TS 4.11.2.2). At least one RHR pump is operating and the breakers of the second pump are verified during these conditions such that performance of this test is only a duplication of existing surveillances. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change does not further degrade the capability of the RHR System to provide decay heat removal since there are alternate Surveillances verifying pump OPERABILITY. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The deletion of this Surveillance does not prevent the RHR System from performing its intended safety function since the Inservice Testing Surveillance is mainly performed to verify pump operation at high pressures which do not exist in MODES 5 and 6. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.43)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.43) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change delays performance of the PORV functional channel test until 12 hours after decreasing to the LTOP enable temperature specified in the PTLR instead of within 31 days prior to entering this condition (current TS 4.16.1.a). The PORVs are only considered as an initiator for a previously analyzed accident with respect to spuriously opening. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The proposed change only provides a short period of time to verify that the PORV is OPERABLE for its LTOP functions since the PORV provides alternate functions, with different setpoints, in higher MODES. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The short period provided to perform the PORV testing ensures that the PORV remains capable of performing its multiple functions through all required MODES. This period of time is consistent with NUREG-1431. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.44)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.44) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows 1 hour to restore compliance for violations of the Reactor Core or RCS Pressure SL in MODES 1 and 2 instead of requiring an immediate shutdown of the plant (current TS 6.7.1.a). Since this change affects the Required Actions following a violation of SLs, this change does not significantly increase the probability of a previously analyzed accident. The proposed change only provides a short period of time to restore compliance before performing a shutdown of the plant in order to limit the potential for additional damage. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The short period provided to restore compliance provides operators with time to stabilize the plant before requiring a shutdown. Therefore, this change does not involve a significant reduction in a margin of safety. This change is also consistent with NUREG-1431 which has been approved by the NRC Staff.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.45)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.45) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change increases the OPERABILITY tolerance for the pressurizer safeties from $\pm 1\%$ to $+ 2.4\%$, -3% (current TS 3.1.1.3.c.). Since the pressurizer safety valve setpoint remains above the normal operating pressure and the PORV setpoint, this change does not significantly increase the probability of a previously analyzed accident. The change has been evaluated with respect to the most limiting pressure transients and shown to be acceptable. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation since the pressurizer safety valve setpoints following testing remain $\pm 1\%$. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The increased OPERABILITY tolerance allows for setpoint drift which has been demonstrated to exist at Ginna Station. The increased tolerances have been analyzed for the most limiting pressure transients with safety limits still being met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.46)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.46) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change increases the fuel enrichment limit from 4.25 weight percent to 5.05 weight percent (current TS 5.3.1.b). The fuel enrichment limit is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The change has been evaluated with respect to fuel handling accidents and shown to be acceptable with respect to offsite doses and 10 CFR 100. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The increased fuel enrichment limit allows for Ginna Station to convert to 18 month cycles. The change has been analyzed and shown that all safety limits are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.47)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.47) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows two SW pumps from the same electrical source to be inoperable for up to 72 hours (current TS 3.3.4.2). The inoperability of one SW train is not considered as an initiator for an accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Systems supported by SW and using the same electrical train as the two SW pumps are currently allowed 72 hours or more to restore one inoperable train. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change only provides consistency within the TS between the SW system and systems which it supports. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.48)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.48) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to "cease operations which may increase the reactivity of the core" (current TS 3.8.2) if the necessary containment penetrations are not isolated during refueling activities. Containment isolation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The required actions for not meeting the containment isolation provisions during refueling is to stop all CORE ALTERATIONS and fuel movement. This action precludes a fuel handling accident for which containment isolation prevents an offsite release. Requiring that all operations which may increase the reactivity of the core is not necessary since containment does not protect against this accident. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. No accident analyses are affected by the removal of this requirement. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.49)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.49) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises the testing requirements of radiation monitors R-11 and R-12 to only require a functional test of the purge valves on a refueling outage basis (every 24 months) versus quarterly (current TS Table 4.1-5, Functional Units #3a and #3b). The radiation monitors are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The radiation monitors actuate the Containment Ventilation Isolation System which is not credited in the accident analyses since it only serves to back up the containment isolation system. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The radiation monitors are not credited in any accident analysis. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.50)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.50) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to isolate containment if the RCS boron concentration is not maintained above 2000 ppm during refueling (current TS 3.6.1.b). Containment isolation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The required actions for not meeting the boron concentration limits is to stop all CORE ALTERATIONS and movement of irradiated fuel. This action precludes a fuel handling accident for which containment isolation prevents an offsite release. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. No accident analyses are affected by the removal of this requirement. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.51)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.51) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change revises current requirements from restoring an inoperable manual AFW or SAFW pump initiation channel within 48 hours to declare the association pump train inoperable (current TS Table 3.5-2, Functional Unit #3.a and #3.f). The manual actuation of the AFW and SAFW pumps is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Since the manual initiation functions only affect one AFW or SAFW pump, entering the LCO for the affected pump is consistent with all other pump operability requirements. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. The change provides consistency within the TS without allowing a pump to be inoperable for a period greater than is currently allowed. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.52)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.52) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change removes the requirement to reduce the high neutron flux trip setpoint to $\leq 85\%$ RTP when power is reduced to $\leq 75\%$ RTP with a misaligned rod (current TS 3.10.4.3.2.b and 3.10.4.3.2.c). The accident analyses are performed assuming that one rod remains fully withdrawn following operation at full power conditions. Since reducing power to $\leq 75\%$ RTP and verifying peaking limits are still maintained must also be performed, the accident analyses remain valid. Therefore, this change does not significantly increase the probability of a previously analyzed accident and does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analyses assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.53)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.53) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change delays the determination of \bar{E} until 31 days after a minimum of 2 EFPDs and 20 days of MODE 1 operation following the reactor being subcritical for ≥ 48 hours (current TS Table 4.1-4, Functional Unit #3). The determination of \bar{E} is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. The allowance of an additional 31 days ensures that a true representative sample is obtained such that the potential for false readings is reduced. The actual value of \bar{E} is not changed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analysis assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

LESS RESTRICTIVE CHANGE CATEGORY (v.b.54)

The proposed changes to the Ginna Station Technical Specifications as discussed in Section D and denoted by Category (v.b.54) do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change allows one diesel generator to be inoperable with no offsite power available for up to 12 hours (current TS 3.7.2.2.d). Since the loss of all offsite power and the failure of a diesel generator are assumptions of most accident analysis, this change does not significantly increase the consequences of an accident. The probability of an accident previously analyzed is not increased since offsite power and the diesel generators only mitigate an accident.
2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The proposed change introduces no new mode of plant operation or changes in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.
3. Operation of Ginna Station in accordance with the proposed change does not involve a significant reduction in a margin of safety. All safety limits and accident analysis assumptions are still met. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

F. ENVIRONMENTAL CONSIDERATION

RG&E has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration as documented in Section D above;

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite since all specifications related to offsite releases are retained, addressed by existing regulations, or relocated to a licensee controlled program subject to the current regulations; and
3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since no new or different type of equipment are required to be installed as a result of this LAR, and the frequency of required testing which may result in radiation exposure is to be optimized consistent with industry practices.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

G. REFERENCES

1. NUREG-1431, *Standard Technical Specifications, Westinghouse Plants*, September 1993.
2. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Conversion to Improved Technical Specifications*, dated February 28, 1994.
3. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 37 to Full-Term Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. 52404)*, dated May 30, 1989.
4. Letter from R.A. Purple, NRC, to E.J. Nelson, RG&E, Subject: *Issuance of Amendment No. 5 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated February 13, 1975.
5. Letter from R.A. Purple, NRC, to E.J. Nelson, RG&E, Subject: *Issuance of Order for Modification of Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated December 27, 1974..
6. NUMARC 93-03, *Writer's Guide for the Restructured Technical Specifications*, February 1993.
7. WCAP-13749, *Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement*, May 1993.
8. Generic Letter 93-05, *Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation*, September 27, 1993.
9. Atomic Industrial Forum (AIF) General Design Criteria (GDC), Issued for comment July 10, 1967.
10. Letter from R.L. Laufer, NRC, to C.A. Schrock, WPS, Subject: *Amendment No. 110 to Facility Operating License No. DPR-43 (TAC No. M88374)*, dated August 3, 1994.
11. WCAP-14040, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, Revision 1, December 1994.
12. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *SEP Topic XV-9, Startup of an Inactive Loop, R.E. Ginna*, dated August 26, 1981.
13. NUREG-0452, *Westinghouse Standard Technical Specifications*.
14. Letter from D.L. Ziemann, NRC, to L.D. White, RG&E, Subject: *Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated July 26, 1979.
15. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *SEP Topics V-10.B, V-11.B, and VII-3*, dated September 29, 1981.

16. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves*, dated April 20, 1981.
17. Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Detection Systems*.
18. NUREG-0821, *Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant*, December 1982.
19. Generic Letter 84-04, *Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops*, February 1, 1984.
20. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Natural Circulation Cooldown, Generic Letter 81-21, R.E. Ginna Nuclear Power Plant*, dated November 22, 1983.
21. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 57 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated December 7, 1993.
22. Federal Register, Volume 60, page 9634, February 21, 1995.
23. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 54 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M77849)*, dated August 30, 1993.
24. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Containment Isolation Boundaries (TAC M77849)*, dated December 21, 1994.
25. Letter from G.E. Lear, NRC, to R.W. Kober, RG&E, Subject: *Containment Purge Technical Specifications, Issuance of Amendment No. 13 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated February 15, 1986.
26. NRC Temporary Inspection 2515/126, *Evaluation of On-Line Maintenance*.
27. Letter from D.M. Crutchfield, NRC, to L.D. White, RG&E, Subject: *Lessons Learned Category "A" Evaluation*, dated February 15, 1986.
28. Letter from J.A. Zwolinski, NRC, to R.W. Kober, RG&E, Subject: *TMI Action Plan Technical Specifications*, dated April 20, 1981.
29. Westinghouse, *Criticality Analysis of the R.E. Ginna Nuclear Power Plant Fresh and Spent Fuel Racks, and Consolidated Rod Storage Canisters*, dated June, 1994.
30. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Change to Technical Specification Instrumentation Requirements. Conversion to Improved Technical Specifications*, dated August 31, 1995.

31. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Change to Technical Specification Instrumentation Requirements, Conversion to Improved Technical Specifications*, dated August 31, 1995.
32. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Diesel Generator Surveillance and Testing*, dated April 23, 1981.
33. NUREG-0944, *Safety Evaluation Report Related to the Full-Term Operating License for R.E. Ginna Nuclear Power Plant*, dated October 1983.
34. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Safety Evaluation for Ginna - SEP Topic VIII-2*, dated June 24, 1981.
35. Letter A.R. Johnson (NRC), to R.C. Mecredy (RG&E), Subject: *Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3*, dated February 24, 1993.
36. Letter from M.B. Fairtile, NRC, to R.W. Kober, RG&E, Subject: *Technical Specifications on Battery Discharge Testing*, dated May 8, 1986.
37. Letter from D.M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: *Safety Evaluation for Ginna - SEP Topic VIII-3A*, dated July 31, 1981.
38. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Issuance of Amendment No. 51 to Operating License No. DPR-18*, dated April 13, 1993.
39. Letter from J.A. Zwolinski, NRC, to R.W. Kober, RG&E, Subject: *Increase of the Spent Fuel Pool Storage Capacity*, dated November 14, 1984.
40. Letter from W.A. Paulson, NRC, to R.W. Kober, RG&E, Subject: *Plant Staff Working Hours and Reporting Requirements for Safety Valve and Relief Valve Failures and Challenges*, dated January 31, 1984.
41. Letter from R.W. Kober, RG&E, to M. Fairtile, NRC, Subject: *Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04)*, dated May 14, 1986.
42. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *NUREG-0737, Item I.A.1.1, Shift Technical Advisor*, dated October 12, 1989.
43. Letter from D.M. Crutchfield, NRC, to L.E. White, RG&E, Subject: *Issuance of Amendment No. 33 to Provisional Operating License No. DPR-18*, dated June 13, 1980.
44. Letter D.M. Crutchfield (NRC) to J.E. Maier (RG&E), *Safety Evaluation for Ginna: SEP Topic VII-6*, dated June 24, 1981.
45. IE Bulletin 79-06A, *Review of Operational Errors and System Misalignments Identified During TMI Incident*.
46. Letter from D.L. Ziemann, NRC, to L.D. White, RG&E, Subject: *Issuance of Amendment No. 27 to Provisional Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant*, dated June 15, 1979.

47. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Generic Letter 88-20*, dated March 15, 1994.
48. WCAP-10271-P-A, Supplement 2, Rev.1, June 1990.
49. Letter from D.M. Crutchfield (NRC) to J. Maier (RG&E), Subject: *Fuel Handling Accident Inside Containment*, dated October 7, 1981.
50. WCAP-13029, *MERITS Program, Phase III, Comments on Draft NUREG-1431, Standard Technical Specifications Westinghouse Plants*, July 1991.
51. WCAP-12159, *MERITS Program, Phase II, Technical Specifications and Bases*, March 1989.
52. WCAP-11618, *MERITS Program, Phase II, Task 5, Criteria Application*, November 1987.
53. ASME, Boiler and Pressure Vessel Code, Section XI.
54. EG&E Report, EGG-NTAP-6175, *In-Service Leak Testing of Primary Pressure Isolation Valves*, February 1983.
55. Letter from V.L. Rooney, NRC, to J.F. Opeka, Northeast Nuclear Energy Company, Subject: *Issuance of Amendment No. 105 (TAC No. M89518)*, dated February 22, 1995.
56. Generic Letter 88-16, *Removal of Cycle-Specific Parameter Limits from Technical Specifications*, dated October 4, 1988.
57. Letter from A.G. Hansen, NRC, to R.E. Link, Subject: *Amendment Nos. 157 and 161 to Facility Operating License Nos. DPR-24 and DPR-27 (TACS M85689 and M85690)*, dated December 8, 1994.
58. Ginna Station LER 95-001, Subject: *Pressurizer Safety Valve Lift Settings Found Above Technical Specification Tolerance During Post-Service Test Due to Setpoint Shifts, Results in Independent Train Being Considered Inoperable*, dated March 6, 1995.
59. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: *Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 3 (TAC No. M80439)*, dated February 24, 1993.
60. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Generic Letter 90-06, Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability" and Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors,"* dated September 15, 1992.
61. Letter from R.E. Smith, RG&E, to C. Stahle, NRC, Subject: *Change P-10 Permissive*, dated December 22, 1988.
62. Letter from C.I. Grimes, NRC, to NSSS Owners Groups, Subject: *Use of Generic Titles in STS,* dated November 10, 1994.

63. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, Subject: *Application for Amendment to Facility Operating License, Implementation of 10 CFR 50, Appendix J, Option B*, dated November 27, 1995.
64. Letter from M. Modes, NRC, to R.C. Mecredy, RG&E, Subject: *10 CFR 50.54 - Quality Assurance Program Change Review*," dated March 22, 1995.

ATTACHMENT B

Marked Up Copy of R.E. Ginna Nuclear Power Plant
Technical Specifications

Included pages:

All pages in Full-Term Operating License and Appendix A to that license up to, and including, Amendment No. 59. These are organized with respect to the proposed new ITS provided in Attachment C.

The following pages have been revised since the May 26, 1995 submittal:

License: 2, 4, and 5

Chapter 1.0: 1.0-5, -8, -9, and -10

Chapter 2.0: None

Chapter 3.0: 3.0-1

Chapter 3.1: 3.1-36

Chapter 3.2: 3.2-1 and 3.2-17a

Chapter 3.3: 3.3-2, -3, -4, -13, -14, -15, -19 through -29, -31, -33 through -37, -50, -52, -54, and -55

Chapter 3.4: 3.4-4, -25, -45, -47, and -55

Chapter 3.5: 3.5-1, and -9

Chapter 3.6: 3.6-5, -6, -9, -10, -11, -29, -30, and -31

Chapter 3.7: 3.7-2, -4, -5, -18, -19, -28, -32 through -36

Chapter 3.8: 3.8-14, -15, and -17

Chapter 3.9: 3.9-1 through -8, -10, -14a, and -14b

Chapter 4.0: 4.0-2, -2a, and -12

Chapter 5.0: 5.0-7, -8, -40, -42, -43, -92, -93, -94, -95, -100



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

FACILITY OPERATING LICENSE

License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the regulations of the Commission set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;
 - B. Construction of the R. E. Ginna Nuclear Power Plant (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-19, as amended, and the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission; *(excepted as exempted from compliance in Section 2.D below)*
 - D. There is reasonable assurance (i) that the facility can be operated at power levels up to 1520 megawatts (thermal) without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the regulations of the Commission;
 - E. The applicant is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;
 - F. The applicant has furnished proof of financial protection that satisfies the requirements of 10 CFR Part 140; and
 - G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.
2. The Provisional Operating License dated September 19, 1969, is superseded by Facility Operating License No. DPR-18 hereby issued to Rochester Gas and Electric Corporation to read as follows:
 - A. This license applies to the R. E. Ginna Nuclear Power Plant, a closed cycle, pressurized, light-water-moderated and cooled reactor, and electric generating equipment (herein referred to as "the

facility") which is owned by the Rochester Gas and Electric Corporation (hereinafter "the licensee" or "RG&E"). The facility is located on the licensee's site on the south shore of Lake Ontario, Wayne County, New York, about 16 miles east of the City of Rochester and is described in license application Amendment No. 6, "Final Facility Description and Safety Analysis Report," and subsequent amendments thereto, and in the application for power increase notarized February 2, 1971, and Amendment Nos. 1 through 4 thereto (herein collectively referred to as "the application").

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses RG&E:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Wayne County, New York, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material or reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as amended, and Commission Safety Evaluations dated November 15, 1976, October 5, 1984, ~~and~~ November 14, 1984, and August 30, 1985

Fuel enrichment
SER

(a) Pursuant to the Act and 10 CFR Part 70, to receive and store four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979);

(b) Pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the licensee's application dated December 14, 1979 (transmitted by letter dated December 20, 1979), as supplemented February 20, 1980 and March 5, 1980;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

RG&E is authorized to operate the facility at steady-state power levels up to a maximum of 1520 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

(a) The licensee shall implement and maintain in effect all fire protection features described in the licensee's submittals referenced in and as approved or modified by the NRC's Fire Protection Safety Evaluation (SE) dated February 14, 1979 and SE supplements dated December 17, 1980, February 6, 1981, June 22, 1981, February 27, 1985 and March 21, 1985 or configurations subsequently approved by the NRC, subject to provision (b) below.

(b) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

67.ii

~~(c) Deleted~~

S.5.10

67.ii

(4) Secondary Water Chemistry Monitoring Program

The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall be described in the plant procedures and shall include:

- (a) Identification of a sampling schedule for the critical parameters and control points for these parameters;
- (b) Identification of the procedures used to measure the values of the critical parameters;
- (c) Identification of process sampling points;
- (d) Procedure for the recording and management of data;
- (e) Procedures defining corrective actions for off control point chemistry conditions; and
- (f) A procedure identifying (i) the authority responsible for the interpretation of the data, and (ii) the sequence and timing of administrative events required to initiate corrective action.

(5) Systems Integrity

S.5.2

67.i

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as reasonably achievable levels. This program shall include the following:

5.5.2

67.i

- (a) Provisions establishing preventive maintenance and periodic visual inspection requirements; and
- (b) Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

5.5.3

(6) Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

- (a) Training of personnel;
- (b) Procedures for monitoring; and
- (c) Provisions for maintenance of sampling and analysis equipment.

D.

67.iv

The facility requires exemptions from certain requirements of ~~10 CFR 50.46(a)(1), 50.48(c)(4), and Appendix J to 10 CFR Part 50.~~ These include: (1) an exemption from 50.46(a)(1), that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K (SER dated April 18, 1978). The exemption will expire upon receipt and approval of revised ECCS calculations; (2) certain exemptions from Appendix J to 10 CFR Part 50 section III.A.4.(a) maximum allowable leakage rate for reduced pressure tests, section III.B.1 acceptable technique for performing local (Type B) leakage rate tests, section III.D.1 scheduling of containment integrated leakage rate tests, and section III.D.2 testing interval for containment airlocks (SER dated March 28, 1978); and (3) an exemption to the scheduler requirements for the alternative shutdown system as set forth in 10 CFR 50.48(c)(4) (NRC letter dated May 10, 1984). The exemption is effective until startup from the 1986 refueling outage. The aforementioned exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, the exemptions are hereby granted pursuant to 10 CFR 50.12.

Delete Appendix J exemptions per 11/27/95 LAR

67.iii

E. Physical Protection - The licensee shall maintain in effect and fully implement all provisions of the following Commission-approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p), which are being withheld from public disclosure pursuant to 10 CFR 73.21:

- E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27827 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Robert Emmet Ginna Nuclear Plant Physical Security Plan," with revisions submitted through August 18, 1987; "Robert Emmet Ginna Nuclear Plant Guard Training and Qualification Plan" with revisions submitted through July 30, 1981; and "Robert Emmet Ginna Nuclear Plant Safeguards Contingency Plan" with revisions submitted through April 14, 1981. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.
- F. This license is effective as of the date of issuance and shall expire at midnight, September 18, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

Attachment:
Appendix A - Technical Specifications

Date of Issuance: December 10, 1984

1. xxiii

1. xxiv

1. xxv

1. xxvi

TECHNICAL SPECIFICATIONS1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 Thermal Power

1.1 The rate that the thermal energy generated by the fuel is accumulated by the coolant as it passes through the reactor vessel.

1.2 Reactor Operating Modes

1.1

Table 1.1-1

Mode	Reactivity $\Delta k/k\%$	Coolant Temperature (°F)
Refueling	≤ -5	$T_{avg} \leq 140$
Cold Shutdown	≤ -1	$T_{avg} \leq 200$
Hot Shutdown	≤ -1	$T_{avg} \geq 540$
Operating	> -1	$T_{avg} \sim 580$

1.3 Refueling

1.1

Table 1.1-1

1.1.1 Any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted.

1-1

1.4

Operable-Operability

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal or emergency electrical power sources (subject to Section 3.0.2), cooling or seal water, lubrication, supports, or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

1.5 Operating

Performing all intended functions in the intended manner.

1.6 Degree of Redundancy (Instrument Channels)

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.7 Instrument Surveillance

1.7.1 Channel Calibration

The adjustment, as necessary, of the channel output so that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The Channel Calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the Channel Functional Test. The Channel Calibration may be performed by any series of sequential, overlapping or total channel steps so that the entire channel is calibrated.

1.7.2 Channel Check

The qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.7.3

Channel Functional Test

- a. Analog channels - the injection of a simulated or source signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated or source signal into the sensor to verify operability including alarm and/or trip function.

1.vii

1.7.4

Source Check

The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.viii

1.8

Containment Integrity

Containment integrity is defined to exist when:

- a. All non-automatic containment isolation valves which are not required to be open during accident conditions are closed and blind flanges are installed where required.
- b. The equipment door is properly closed and sealed.
- c. At least one door in each personnel air lock is properly closed and sealed.
- d. All automatic containment isolation valves are operable, secured in the closed position or isolated by closed manual valves or flanges as permitted by Limiting Conditions for Operation.
- e. The containment leakage satisfies Technical Specification 4.4.

1.1x

Relocated to bases for 3.6.1 and 3.6.2

1.9

Quadrant Power Tilt

1.1

The ratio of highest average nuclear power in any quadrant to the average nuclear power in the four quadrants. If one excor detector is out of service, the three inservice units are used in computing the average.

1.10

Hot Channel Factors

F_Q , Heat Flux Hot Channel Factor, is defined as the maximum local heat flux in the surface of a fuel rod divided by the average fuel rod heat flux allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N , Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density assuming nominal fuel pellet and rod dimensions.

F_Q^E , Engineering Heat Flux Hot Channel factor,, is defined as the ratio between F_Q and F_Q^N and is the allowance on heat flux required for manufacturing tolerances.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

(1. x)

(1. xi)

~~1.11 (DELETED)~~

1.12

Frequency Notation

The frequency notation specified for the performance of surveillance requirements shall correspond to the intervals defined below.

<u>Notation</u>	<u>Frequency</u>
S, Each Shift	At least once per 12 hours
D, Daily	At least once per 24 hours
Twice per week	At least once per 4 days and at least twice per 7 days
W, Weekly	At least once per 7 days
B/W, Biweekly	At least once per 14 days
M, Monthly	At least once per 31 days
B/M, Bimonthly	At least once per 62 days
Q, Quarterly	At least once per 92 days
SA, Semiannually	At least once per 6 months
A, Annually	At least once per 12 months
R	At least once per 18 months
S/U	Prior to each startup
N.A.	Not Applicable
P	Prior to each startup if not done previous week
PR	Within 12 hours prior to each release

1.4

(1.xii)

1.13

(1.xiii)

Offsite Dose Calculation Manual (ODCM)

The ODCM is a manual containing the methodology and parameters to be used for calculating the offsite



1.12

Frequency Notation

The frequency notation specified for the performance of surveillance requirements shall correspond to the intervals defined below.

<u>Notation</u>	<u>Frequency</u>
S, Each Shift	At least once per 12 hours
D, Daily	At least once per 24 hours
Twice per week	At least once per 4 days and at least twice per 7 days
W, Weekly	At least once per 7 days
B/W, Biweekly	At least once per 14 days
M, Monthly	At least once per 31 days
B/M, Bimonthly	At least once per 62 days
Q, Quarterly	At least once per 92 days
SA, Semiannually	At least once per 6 months
A, Annually	At least once per 12 months
R	At least once per 18 months
S/U	Prior to each startup
N.A.	Not Applicable
P	Prior to each startup if not done previous week
PR	Within 12 hours prior to each release

1.4

1.xii

1.13

Offsite Dose Calculation Manual (ODCM)

The ODCM is a manual containing the methodology and parameters to be used for calculating the offsite

See Chapter 5.02

See Chapter
5.0

doses due to liquid and gaseous radiological effluents, in calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.14 Process Control Program (PCP)

The PCP is a manual outlining the method for processing wet solid wastes and for solidification of liquid wastes. It shall include the process parameters and evaluation methods used to assure meeting the requirements of 10 CFR Part 71 prior to shipment of containers of radioactive waste from the site.

1.xiv

1.15 Solidification

Solidification shall be the conversion of radioactive wastes from liquid systems to a homogeneous solid.

1.xv

1.16 Purge-Purging

Purge or purging is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confined space.

1.xvi

1.17 Venting

Venting is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air is not provided or required.

1.xvii

1.18 Dose Equivalent I-131

1.1

The dose equivalent I-131 shall be that concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those for the adult thyroid dose via inhalation, contained in NRC Regulatory Guide 1.109 Rev. 1 October 1977.

1.19 Reportable Event

1.xix

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

1.20 Canisters Containing Consolidated Fuel Rods

1.xx

Canisters containing consolidated fuel rods are stainless steel canisters containing the fuel rods of no more than two fuel assemblies which have decayed at least five years and are capable of being stored in a storage cell of the spent fuel pool.

1.21 Shutdown Margin

1.1

1.xx1

Shutdown margin shall be the amount of reactivity by which the reactor is subcritical, or would be subcritical from its present condition assuming all rod cluster control assemblies (shutdown and control) are fully inserted except for the single rod cluster control assembly of highest reactivity worth which is assumed to be fully withdrawn, and assuming no changes in xenon or boron concentration.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Reactor Core

Applicability:

2.1

Applies to the limiting combinations of thermal power, reactor coolant system pressure and coolant temperature during operation ← MODES 1 and 2

Objective:

To maintain the integrity of the fuel cladding.

Specification:

The combination of thermal power level, coolant pressure and coolant temperature shall not exceed the limits shown in Figure 2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

2.1.1

Basis:

~~To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer; wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable~~

parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and/or WRB-1 DNB correlation. These DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. A minimum value of the DNB ratio, MDNBR, is specified so that during steady state operation, normal operational transients and anticipated transients, there is a 95% probability at a 95% confidence level that DNB will not occur. (1) The curves of Figure 2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which this minimum DNB value is satisfied. The area of safe operation is below these lines.

Since it is possible to have somewhat greater enthalpy rise hot channel factors at part power than at full power due to the deeper control bank insertion which is permitted at part power, a conservative allowance has been made in obtaining the curves in Figure 2.1-1 for an increase in $F_{\Delta H}^N$ with decreasing power levels. Rod withdrawal block and load runback occurs before reactor trip set points are reached.

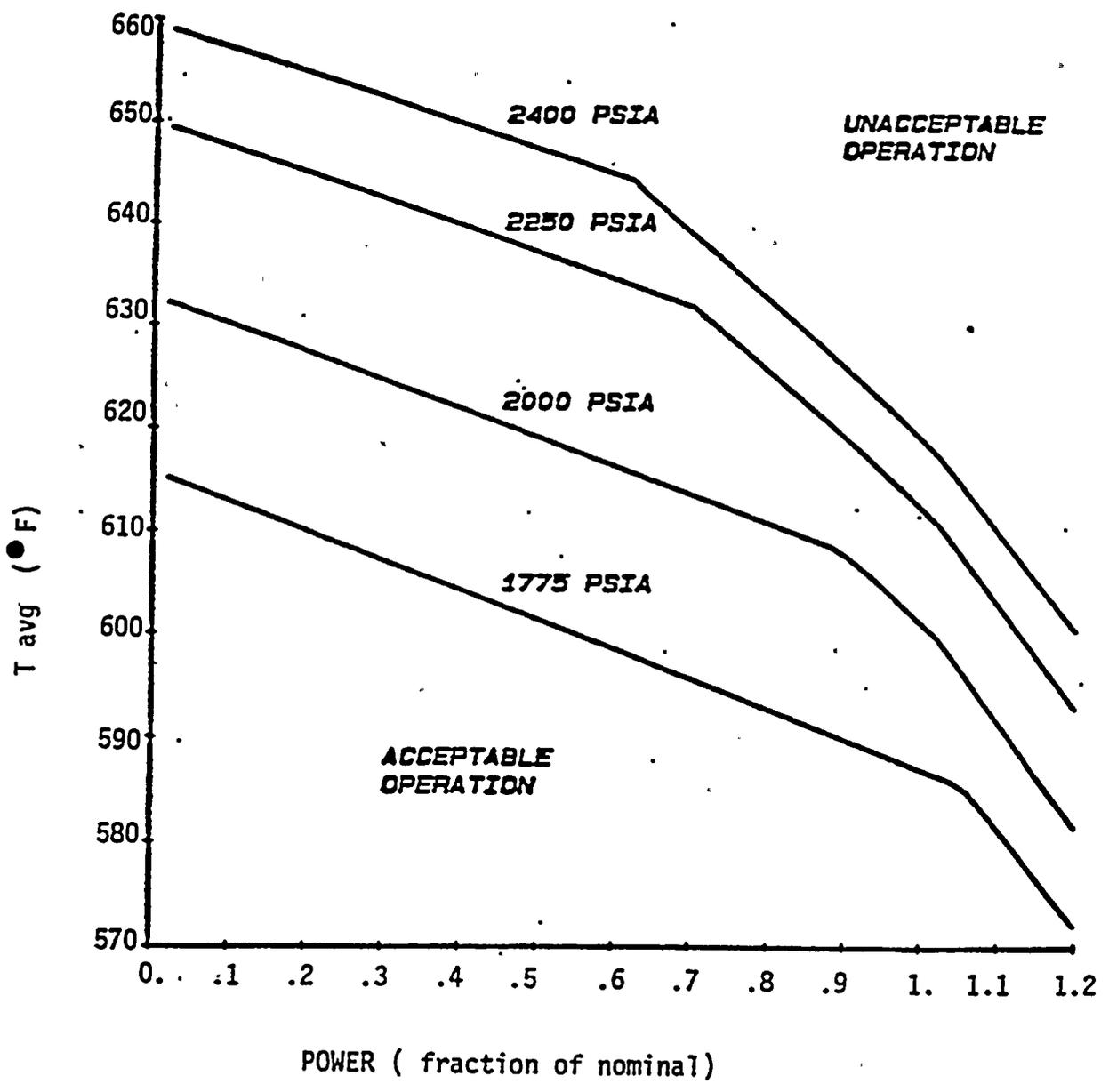
The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure and thermal power level that

would result in there being less than a 95% probability at a 95% confidence level that DNB would not occur. (2)

(1) FSAR, Section 3.2.2

(2) Safety Evaluation for R.E. Ginna Transition to 14 x 14 Optimized Fuel Assemblies, Westinghouse Electric Corporation, November 1983.

2.1.1-1
FIGURE ~~2.1.1~~
CORE DNB SAFETY LIMITS
2 LOOP OPERATION



~~2.1.4~~

2.2 Safety Limit - Reactor Coolant System Pressure

Applicability:
 Applies to the limit on Reactor Coolant System pressure.

Objective:
 To maintain the integrity of the Reactor Coolant System.

2.1.2

Specification:
 The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

3.i

Basis:
 The Reactor Coolant System⁽¹⁾ serves as a barrier preventing radio-nuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system and fuel cladding is assured. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.⁽²⁾

The settings of the power-operated relief valves (2335 psig), the reactor high pressure trip (2385 psig), and the safety valves (2485 psig) have been established to assure never reaching the Reactor Coolant System pressure safety limit. The initial hydrostatic test was conducted at 3110 psig to assure the integrity of the Reactor Coolant System.

References:

- (1) FSAR, Section 4
- (2) FSAR, Section 4.3

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

2.2
SS.i

a. The provisions of 10 CFR 50.36(c)(1)(i)(A) shall be complied with immediately.

SS.ii

b. The Safety Limit violation shall be reported to the Senior Vice President, Customer Operations*, to the offsite review function, and to the NRC immediately.

SS.iii

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the onsite review function. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

SS.iv

d. The Safety Limit Violation Report shall be submitted to the NRC, the offsite review function, and the Senior Vice President, Customer Operations* within two weeks of the violation.

SS.ii

SS.iv

* An alternate title may be designated for this position in accordance with 10 CFR 50.54(a)(3). All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

3.0 Add 100 3.0.1
3.0 Add 100 3.0.2

LIMITING CONDITION FOR OPERATION

APPLICABILITY

3.0.1

In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, within 1 hour action shall be initiated to place the unit in at least hot shutdown within the next 6 hours (i.e., a total of seven hours), and in at least cold shutdown within the following 30 hours (i.e., a total of 37 hours) unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a mode in which the specification is not applicable. If the action statement corresponding to the Limiting Condition for Operation that was exceeded contains time limits to hot and cold shutdown that are less than those specified above, these more limiting time limits shall be applied. Exceptions to these requirements shall be stated in the individual specifications.

100 3.0.3
5.vii
5.viii

3.0.2

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its preferred power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided:

100 3.0.4
5.vi

5.iv Add 100 3.0.4
5.v Add 100 3.0.5

3.0.7

5.vi

(1) its corresponding preferred or emergency power source is operable; and (2) all of its redundant system(s), subsystems(s), train(s), component(s) and device(s) are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 1 hour, the unit shall be placed in at least hot shutdown within the next 6 hours, and in at least cold shutdown within the following 30 hours. This specification is not applicable in cold shutdown or refueling modes.

Basis

Specification 3.0.1 delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.3.2 requires two Containment Spray Pumps to be operable and provides explicit action requirements if one spray pump is inoperable. Under the terms of Specification 3.0.1, if both of the required Containment Spray Pumps are inoperable, the unit is required to be in at least hot shutdown within the following 6 hours and in at least cold shutdown in the next 30 hours. These time limits apply because the time limits for one spray pump inoperable (6 hours to hot shutdown, wait 48 hours then 30 hours to cold shutdown) are less limiting. As a further example, Specification 3.3.1 requires each Reactor Coolant System accumulator to be operable and provides explicit action requirements if one accumulator is inoperable. Under the terms of Specification 3.0.1, if more than one accumulator is inoperable, within 1 hour action shall be initiated to place the unit in at least hot shutdown within 6 hours and cold shutdown within an additional 30 hours. The time limit of 6 hours

to hot shutdown and 30 hours to cold shutdown do not apply because the time limits for 1 accumulator inoperable are more limiting. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate action statement.

Specification 3.0.2 delineates what additional conditions must be satisfied to permit operation to continue, consistent with the action statements for power sources, when a preferred or emergency power source is not operable. It allows operation to be governed by the time limits of the action statement associated with the Limiting Condition for Operation for the preferred or emergency power source, not the individual action statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its preferred or emergency power source.

For example, Specification 3.7.2.1.a requires in part that two emergency diesel generators be operable. The action statement provides for a maximum out-of-service time when one emergency diesel generator is not operable. If the definition of operable were applied without consideration of Specification 3.0.2, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.2 permit the time limits for continued operation to be consistent with the action statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding

preferred power source must be operable, and all redundant systems, subsystems, trains, components, and devices must be operable, or otherwise satisfy Specification 3.0.2 (i.e., be capable of performing their design function and have at least one preferred or one emergency power source operable). If they are not satisfied, shutdown is required in accordance with this specification.

27.i — Add SR 3.0.1

27.iii — Add SR 3.0.3

27.iv — Add SR 3.0.4

4.0 SURVEILLANCE REQUIREMENTS

SR 3.0.2
27.ii Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 Operational Safety Review

Applicability:

Applies to items directly related to safety limits and limiting conditions for operation:-

Objective:

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

- 4.1.1 Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.
- 4.1.2 Equipment and sampling tests shall be conducted as specified in Table 4.1-2 and 4.1-4.
- 4.1.3 Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check and channel calibration operations at the frequencies shown in Table 4.1-3.
- 4.1.4 Each radioactive effluent monitoring instrumentation channel shall be demonstrated operable by performing the channel check, source check, channel functional test, and channel calibration at the frequency shown in Table 4.1-5.

See all other chapters

3.1.3 Minimum Conditions for Criticality

8.IV

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at a temperature below 500°F, and if the moderate temperature coefficient is more positive than

LCO 3.1.3
LCO 3.1.8

8.V

- a. 5 pcm/°F (below 70 percent of rated thermal power)
- b. 0 pcm/°F (at or above 70 percent of rated thermal power)

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit line shown on Figure 3.1-1 of these specifications.

See Chapter 3.4

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

Basis

~~Previous safety analyses have assumed that for Design Basis Events (DBE) initiated from the hot zero power or higher power condition, the moderator temperature coefficient (MTC) was either zero or negative. (1)(2) Beginning in Cycle 14, the safety analyses have assumed that a maximum MTC of +5 pcm/°F can exist up to 70% power. Analyses have shown that the design criteria can be satisfied for the DBE's with this assumption. (3) At greater than 70% power the MTC must be zero or negative.~~

~~The limitations on MTC are waived for low power physics tests to permit measurement of the MTC and other physics design parameters of interest. During these tests special operating precautions will be taken.~~

See
Chapter
3.4

The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase in moderator temperature or a decrease of coolant pressure.

Reference

- ~~(1) FSAR Table 3.2.1-1~~
- ~~(2) FSAR Figure 3.2.1-8~~
- ~~(3) Safety Evaluation for R.E. Ginna Transition to 14 x 14 Optimized Fuel Assemblies, Westinghouse Electric Corporation, November 1983.~~

3.2 Chemical and Volume Control System

Applicability

Applies to the operational status of the chemical and volume control system.

Objective

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation.

Specification

3.2.1

12.ii

During cold shutdown or refueling with fuel in the reactor there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

3.2.1.1

With this flow path unavailable, immediately suspend all operations involving core alterations or positive reactivity changes and return a flow path to operable status as soon as possible.

3.2.2

12.iii

When the reactor is above cold shutdown, two boron injection flow paths shall be operable with one operable charging pump for each operable flow path, and one operable boric acid transfer pump for each operable flow path from the boric acid storage tank(s).

3.2.3

12.iv

If required by specification 3.2.2 above, the Boric Acid Storage Tank(s) shall satisfy the concentration, minimum volume and solution temperature requirements of Table 3.2-1.

3.2.4

(12.iii)

With only one of the required boron injection flow paths to the RCS operable, restore at least two boron injection flow paths to the RCS to operable status within 72 hours, or within the next 6 hours be in at least hot shutdown and borated to a shutdown margin equivalent to at least 2.45% delta k/k at cold, no xenon conditions. If the requirements of 3.2.2 are not satisfied within an additional 7 days, then be in cold shutdown within the next 30 hours.

3.2.5

See
Chapter
3.4

Whenever the RCS temperature is greater than 200°F and is being cooled by the RHR system and the over-pressure protection system is not operable, at least one charging pump shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull-stop position.

Table 3.2-1

Boric Acid Storage Tank(s)

Minimum-Volume-Temperature-Concentration⁽²⁾

Concentration ppm boron	Minimum Volume gal.	Minimum Solution Temperature °F
4700 to less than 5000	8400	40
5000 to less than 6000	7800	52
6000 to less than 7000	6400	62
7000 to less than 8000	5400	70
8000 to less than 9000	4700	78
9000 to less than 10000	4200	85
10000 to less than 11000	3800	91
11000 to less than 12000	3500	97
12000 to less than 13000	3200	103
13000 to less than 14000	3000	108
14000 to less than 15000	2700	113
15000 to less than 16000	2500	118
16000 to less than 17000	2400	123
17000 to less than 18000	2200	127
18000 to less than 19000	2100	131
19000 to less than 20000	2000	137
20000 to less than 21000	1900	140
21000 to less than 22000	1800	143
22000 to less than 23000	1800	145

Basis

The chemical and volume control system provides control of the reactor system boron inventory.⁽¹⁾ This is normally accomplished by using one or more charging pumps in series with one of the two boric acid transfer pumps.

Above cold shutdown conditions, a minimum of two of four boron injection flowpaths are required to insure single functional capability in the event that an assumed single active failure renders one of the flow paths inoperable. The boration volume available through any flow path is sufficient to provide the required shutdown margin at cold conditions from any expected operating condition and to compensate for shrinkage of the primary coolant from the cooldown process. The maximum volume requirement is associated with boration from just critical, hot zero power, peak xenon with control rods at the insertion limit, to cold shutdown with single reactor coolant loop operation. This requires 26,000⁽²⁾ gallons of 2000 ppm borated water from the refueling water storage tank or the concentrations and volumes of borated water specified in Table 3.2-1 from the boric acid storage tanks. Two boric acid storage tanks are available. One of the two tanks may be out of service provided the required volume of boric acid is available to the operable flow paths.

Above cold shutdown, two of the following four flow paths must be operable with one operable charging pump for each operable flow path, and one operable boric acid transfer pump for each operable flow path from the boric acid storage tanks.

- (1) Boric acid storage tanks via one boric acid transfer pump through the normal makeup (FCV 110A) flow path to the suction of the charging pumps.
- (2) Boric acid storage tanks via one boric acid transfer pump through the emergency boration flow path (MOV 350) to the suction of the charging pumps.
- (3) Refueling water storage tank via gravity feed through AOV 112B to the suction of the charging pumps.

refueling water storage tank via gravity feed through manual bypass valve 358 to the suction of the charging pumps. 3.1-7

Available flow paths from the charging pumps to the reactor coolant system include the following:

- (1) Charging flow path through AOV 392A to the RCS Loop B hot leg.
- (2) Charging flow path through AOV 294 to the RCS Loop B cold leg.
- (3) Seal injection flow path to the reactor coolant pumps.

The rate of boric acid injection must be sufficient to offset the maximum addition of positive reactivity from the decay of xenon after a trip from full power. This can be accomplished through the operation of one charging pump at minimum speed with suction from the refueling water storage tank. Also the time required for boric acid injection allows for the local alignment of manual valves to provide the necessary flow paths.

The quantity of boric acid specified in Table 3.2-1 for each concentration is sufficient at any time in core life to borate the reactor coolant to the required cold shutdown concentration and provide makeup to maintain RCS inventory during the cooldown. The temperature limits specified on Table 3.2-1 are required to maintain solution solubility at the upper concentration in each range. The temperatures listed on Table 3.2-1 are taken from Reference (4). An arbitrary 5°F is added to the Reference (4) for margin. Heat tracing may be used to maintain solution temperature at or above the Table 3.2-1 limits. If the solution temperature of either the flow path or the borated water source is not maintained at or above the minimum temperature specified, the affected flow path must be declared inoperable and the appropriate actions specified in 3.2.4 followed.

Placing a charging pump in pull-stop whenever the reactor coolant system temperature is $\geq 200^\circ\text{F}$ and is being cooled by RHR without the overpressure protection system operable will prevent inadvertent overpressurization of the RHR system should letdown be terminated.⁽³⁾

References:

- (1) UFSAR Section 9.3.4.2
- (2) RG&E Design Analysis DA-NS-92-133-00 "BAST Boron Concentration Reduction Technical Specification Values" dated Dec. 14, 1992
- (3) L.D. White, Jr. letter A. Schwencer, NRC, Subject: Reactor Vessel Overpressurization, dated February 24, 1977

(4) ~~Kerr-McGee Chemical Corp. Bulletin 0151 "Boric Acid - Technical Grades" dated 5/84~~

~~3.10 Control Rod and Power Distribution Limits~~

~~Applicability~~

~~Applies to the operation of the control rods and power distribution limits.~~

~~Objective~~

~~To ensure (1) core subcriticality after a reactor trip, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.~~

~~Specification~~

3.10.1 Control Rod Insertion Limits.

20.i

3.10.1.1 When the reactor is subcritical prior to startup, the hot shutdown margin shall be at least that shown in

LCO 3.1.1
LCO 3.1.2

20.ii
20.iii

Figure 3.10-2. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (547°F) if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon or boron.

specified in the COLR.

~~Amendment No. 10, 24
March 30, 1976~~

3.10.1.2
LCD 3.1.5
LCD 3.1.8

When the reactor is critical except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn (indicated position).

20.iv

low power

20.iv

low power

3.10.1.3
LCD 3.1.6
LCD 3.1.8

When the reactor is critical, except for physics tests and control rod exercises, each group of control rods shall be inserted no further than the limits shown by the lines on Figure 3.10-1 and moved sequentially with a 100 (+5) step (demand position) overlap between successive banks.

Specified in the COLR.

20.v

Relocate to COLR

3.10.1.4
LCD 3.1.5
LCD 3.1.6

During control rod exercises indicated in Table 4.1-2, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.1.5

~~During measurement of control rod worth and shutdown margin, the shutdown margin requirement, Specification 3.10.1.1, need not be observed provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion. Each full length control rod not fully inserted, that is, the rods available for trip insertion, shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position (indicated) within 24 hours prior to reducing the shutdown margin to less than the limits of Specification 3.10.1.1. The position of each full length rod not fully inserted, that is, available for trip insertion, shall be determined at least once per 2 hours.~~

20.vi

3.10.2 Power Distribution Limits and Misaligned Control Rod

3.10.2.1 The movable detector system shall be used to measure power distribution after each fuel reloading prior to operation of the plant at 50% of rated power to ensure that design limits are not exceeded.

If the core is operating above 75% power with one excore nuclear channel out of service, then the quadrant to average power tilt ratio shall be determined once a day by at least one of the following means:

- a. Movable detectors
- b. Core-exit thermocouples

See Chapter 3.2

3.10.2.2 Power distribution limits are expressed as hot channel factors. At all times, ~~except during low power physics tests~~ the hot channel factors must meet the following limits:

20.vii

$$F_Q(Z) = (2.32/P)*K(Z) \quad \text{for } P \geq .5$$

$$F_Q(Z) = 4.64*K(Z) \quad \text{for } P \leq .5$$

$$F_{\Delta H}^N = 1.66 [1 + .3(1-P)] \quad \text{for } 0 \leq P \leq 1.00$$

Where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q shall be increased by three percent to yield F_Q . If the measured F_Q or $F_{\Delta H}^N$ exceeds the limiting value, with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced one percent for each percent with $F_{\Delta H}^N$ or F_Q exceeds the limiting value, whichever is more restrictive. If the hot channel factors cannot be reduced below the

See Chapter 3.2

limiting values within one day, the Overpower ΔT trip setpoint and the Overtemperature ΔT setpoint shall be similarly reduced.

20.viii

3.10.2.3 ~~Except for physics tests,~~ If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12, then within two hours:

see Chapter 3.2

- a. Correct the situation, or
- b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or
- c. Limit power to 75% of rated power.

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, within 2 hours either reduce the quadrant to average power tilt ratio to within its limit or reduce power to less than 50% of rated power. Within an additional 4 hours, either reduce the ratio to within its limit or be at hot shutdown. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip setpoint is reduced by 50%.

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that

the hot channel factor limits of Specification 3.10.2.2 are met.
The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by linear interpolation using the most recent measured value and the predicted value at the end of the cycle life.

3.10.2.7

See Chapter 3.2

20.1x

3.10.2.8

Except during ~~physics tests~~, control rod exercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within ±5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

See Chapter 3.2

20.1x

3.10.2.9

Except during ~~physics tests~~, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

See Chapter 3.2

2D. IX

3.10.2.10 Except during ~~physics tests~~, control rod exercises, or excore calibration, at a power level less than or equal to 90 percent of rated power:

- a. The indicated axial flux difference may deviate from its $\pm 5\%$ target band for a maximum of one hour (cumulative) in any 24 hour period, however, the flux difference shall not exceed an envelope bounded by -11 percent and $+11$ percent at 90% power and increasing by -1 percent and $+1$ percent for each 2 percent of rated power below 90% power.
- b. If Specification 3.10.2.10a is violated, then the reactor power shall be immediately reduced to no greater than 50% power.
- c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

See
Chapter
3.2

3.10.2.11 A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period. One half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power.

3.10.2.12 When the reactor is critical and thermal power is less than or equal to 90% of rated power, an alarm is provided to indicate when the axial flux difference has been outside the target band for more than one hour (cumulative) out of any 24 hour period. In addition, when thermal power is greater than 90% of rated power, an alarm is provided to indicate when the axial flux difference is outside the target band. If either alarm is out of service, the flux difference shall be logged hourly for the first 24 hours the alarm is out of service and half-hourly thereafter.

see
Chapter
3.2

3.10.3 Control Rod Drop Time

3.10.3.1 While critical, the individual full length (shutdown and control) rod drop time from the fully withdrawn position (indicated) shall be less than or equal to 1.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

SR 3.1.4.4

20.x

- a. T_{avg} greater than or equal to $540^{\circ}F$ and
b. All reactor coolant pumps operating.

500 °F

3.10.3.2: With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to criticality.

SR 3.1.4.4

20.xi

3.10.4 Control Rod Group Height

3.10.4.1 While critical, and except for physics testing, all full length (shutdown and control) rods shall be operable and positioned within ± 12 steps (indicated position) of their group step counter demand position.

LCO 3.1.4

LCO 3.1.8

low power

3.10.4.2 With any ~~full-length rod inoperable due to being~~
 LCD 3.1.4 ~~immovable as a result of excessive friction or mechanical~~
 (20.xii) ~~interference or known to be untripable, determine that~~
 the shutdown margin requirement of Specification
 3.10.1.1 is satisfied within 1 hour and be in hot
 shutdown within 6 hours.

3.10.4.3 With one ~~full-length rod inoperable due to causes~~
 LCD 3.1.4 ~~other than addressed by 3.10.4.2, above, or misaligned~~
 (20.xii) ~~from its group step counter demand position by more~~
 than ± 12 steps (indicated position), operation may
 continue provided that within one hour either:

3.10.4.3.1 The rod is restored to operable status within the
 LCD 3.1.4 above alignment requirements, or

(20.xiii) 3.10.4.3.2 The ~~rod is declared inoperable and the~~ shutdown margin
 LCD 3.1.4 requirement of Specification 3.10.1.1 is satisfied.
 Operations may then continue provided either:

(20.xiv) LCD 3.1.4 Bases a. The remainder of the rods in the group with the
 inoperable rod are aligned to the same indicated
 position as the inoperable rod within one hour, while
 maintaining the limit of Specification 3.10.1.3: or

LCD 3.1.4 b. The power level is reduced to less than or equal
 to 75% of rated power within the next one hour,
 and ~~the high neutron flux trip setpoint is reduced~~
~~to less than or equal to 85% rated power within~~
~~the next four hours (total of six hours) and the~~
 following evaluations are performed:

(20.xvi) (i) The shutdown margin requirement of Specification
 3.10.1.1 is determined at least once per 12 hours.

LCD 3.1.4

(ii) A power distribution map is obtained from the movable incore detectors and $F_Q(2)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours.

LCD 3.1.4

20.XV

(iii) A reevaluation of each accident analysis ~~of Table 3.10.1~~ is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

~~c. If power has been restricted in accordance with (b) above, then following completion of the evaluation identified in (b), the power level and high neutron flux trip setpoint may be readjusted based on the results of the evaluation provided the shutdown margin requirement of Specification 3.10.1.1 is determined at least once per 12 hours.~~

20.XVI

3.10.4.4
LCD 3.1.4

20.XVII

With two or more full length rods inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in hot shutdown within 6 hours.

3.10.5

Control Rod Position Indication Systems

3.10.5.1

LCD 3.1.7

20.XVIII

While critical, the rod position indication system and the step counters shall be operable and capable of determining the control rod positions within ± 12 steps.

3.10.5.2
LCD 3.1.7

With a maximum of one rod position indication per bank inoperable either:

20.XIX

- a. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps (demand position) in one direction since the last determination of the rod's position, or
- b. Reduce the power to less than 50% of rated power within 8 hours.

3.10.5.3
LCD 3.1.7

With a maximum of one step counter per bank inoperable either:

- a. Verify that position indication for each rod of the affected bank is operable and that the rods of the bank are at the same indicated position at least once per 8 hours, or
- b. Reduce the power to less than 50% of rated power within 8 hours.

Basis

~~The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation the shutdown groups are fully withdrawn~~

and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition. The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis.⁽¹⁾ In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors.

The lines shown on Figure 3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end-of-cycle life and is based on the value used in analysis of the hypothetical steam break accident. Early in cycle life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to that which is required at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-3 has been determined from extensive analyses considering operating maneuvers consistent with the Technical Specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope

demonstrate compliance with the Final Acceptance Criteria limit for Emergency Core Cooling Systems. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading pattern. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which might, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these

conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 25 steps from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as described in Specification 3.10.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution limits which are given in terms of flux difference limits and control bank insertion limits are observed. Flux difference is $q_T - q_B$ as defined in Specification 2.3.1.2d.

The permitted relaxation in $F_{\Delta H}^N$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10, F_Q is arbitrarily limited for $P < 0.5$ (except for lower power physics tests).

The limits on axial power distribution specified to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium

value of Axial Offset (Axial Offset = ΔI /fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies primarily with burnup. The technical specifications on power distribution assure that the F_Q upper bound envelope of 2.32 times Figure 3.10-3 is not exceeded and xenon distributions are not developed which, at a later time, could cause greater local power peaking even though the flux difference is then within the limits.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with control Bank D more than 190 steps (indicated position) withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference.

Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 percent ΔI is permitted from the indicated reference value.

During periods where extensive load following is

~~required, it may be impossible to establish the required
core conditions for measuring the target flux difference
every month. For this reason, two methods are~~

~~3.10-14a~~

~~Amendment No. 22~~

permissible for updating the target flux difference. Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures

that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent of each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation. The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in F_Q^N and $F_{\Delta F}^N$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction on peaking factors.

The two hours in 3.10.2.3 are acceptable since complete rod misalignment (full-length control rod 12

- feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If instead of determining the hot channel factors, the operator decides to reduce power, the specified 75% power maintains the design margin to core safety limits for up to 1.12 power tilt, using the 2 to 1 ratio. Reducing the overpower trip set point ensures that the protection system basis is maintained for sustained plant operation. A tilt ratio of 1.12 or more is indicative of a serious performance anomaly and a plant shutdown is prudent. The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 540°F and with both reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The various control rod banks (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the

demand position of the banks and a microprocessor rod position indication (MRPI) system which indicates the actual rod position. The digital counters are known as the step counters.

Operability of the control rod position indication is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. The 12 step permissible demand to indicated misalignment and the 0 step rod to rod indicated misalignment ensures that the 25 step misalignment assumed in the safety analysis is met. The MRPI system displays the position of all rods on a CRT. A failure of the CRT would result in loss of position indication of the rods even though the MRPI system is still operable. Since the MRPI system also transmits rod position information to the Plant Process Computer System (PPCS), the PPCS can be used for rod position indication until the CRT is made operable.

The action statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a

restriction in power; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

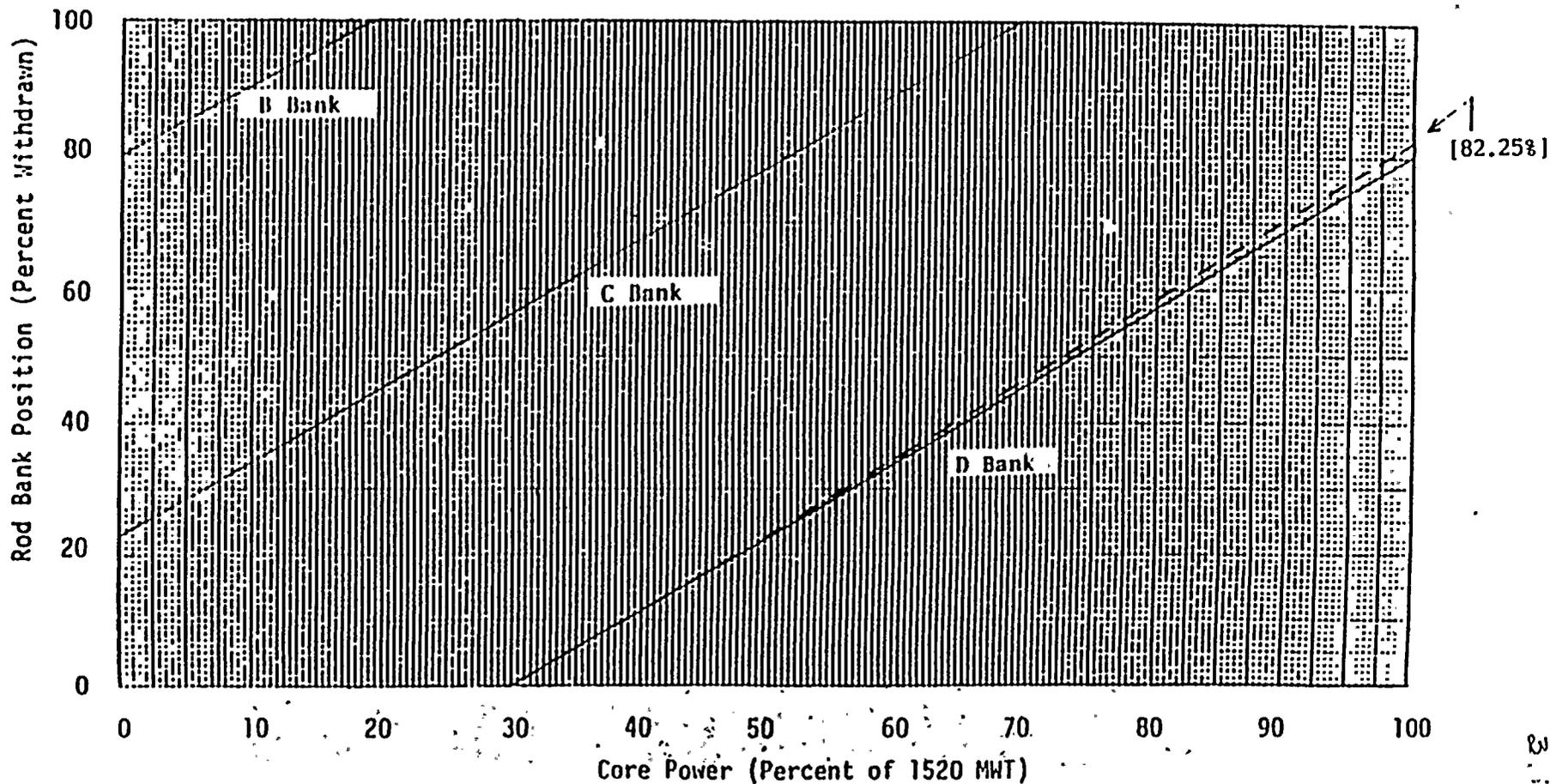
References:

- (1) Updated Final Safety Analysis Report (UFSAR)
Section 4.2.

FIGURE 3.10-1
CONTROL ROD INSERTION LIMITS VERSUS CORE POWER
FOR BOL THROUGH EOL.

20. V
Figures Relocated
to the COLR

[--- Dashed line is applicable only for Cycle 19 with cycle burnup greater than .5250 MWD/MTU providing Cycle 18 burnup has exceeded 12150 MWD/MTU.]



3.10-21

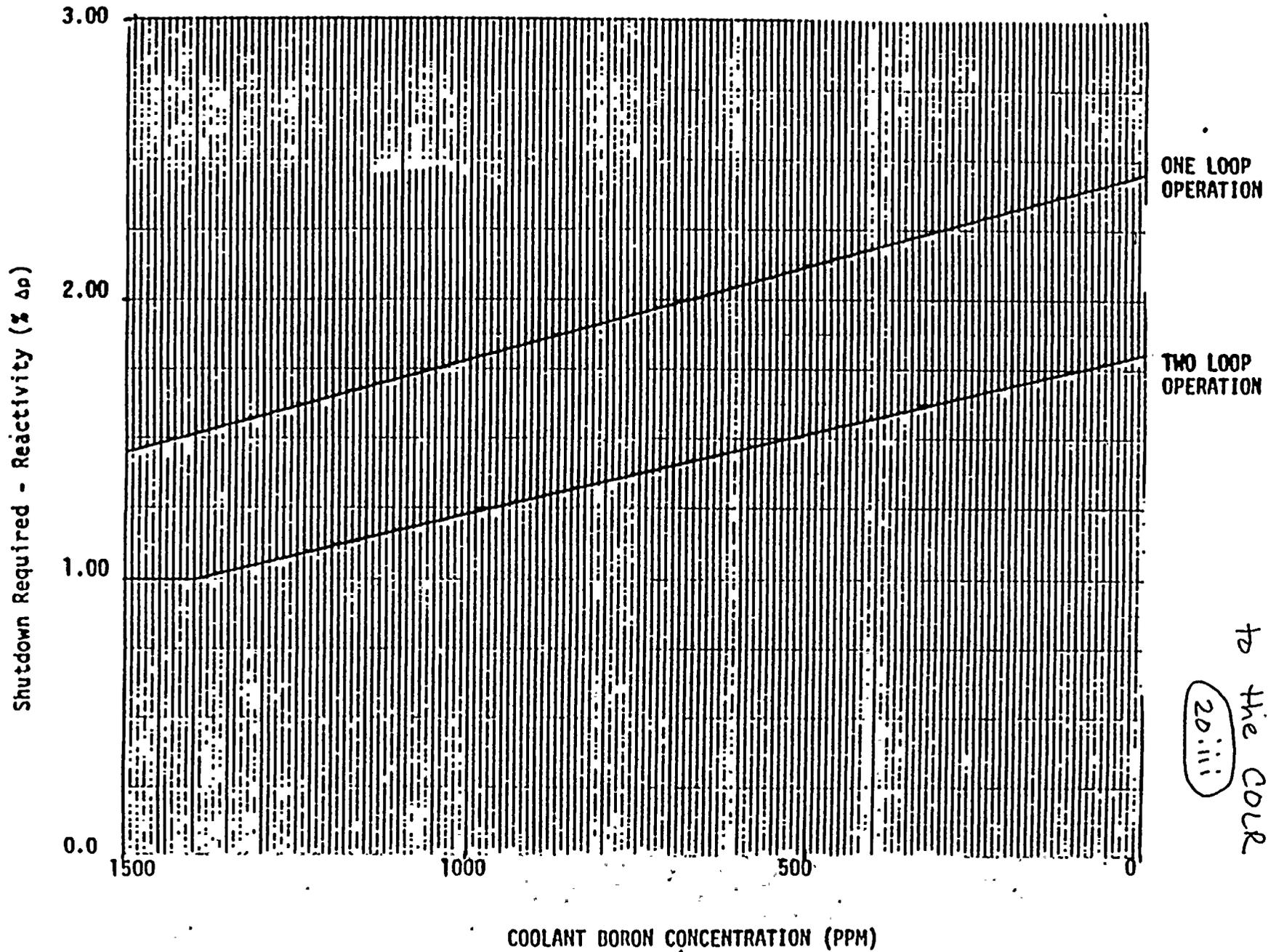


Figure Relocated
to the COR
20.111

COOLANT BORON CONCENTRATION (PPM)

REQUIRED SHUTDOWN MARGIN

FIGURE 3.10-2

NORMALIZED AXIAL DEPENDENCE FACTOR FOR

F_Q VS. ELEVATION

FIGURE 3.10-3

SEE Chapter
3.2

1.2500

1.0000

0.7500

0.5000

0.2500

0.0

0.0

2.000

4.000

6.000

8.000

10.000

12.000

CORE HEIGHT (FT)

TOTAL F_Q

2.320

CORE HEIGHT

$K(Z)$

0.000

1.000

6.000

1.000

10.800

0.940

12.000

0.647

3.10-22
(Z)

20. XV

Table 3.10-1ACCIDENT ANALYSIS REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE CONTROL ROD

Rod Insertion Characteristics

Rod Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks
In Large Pipes Which Actuates The Emergency Core Cooling System

Rod Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant
Accident)

Steam Line Break

Rod Ejection

TABLE 4.1-1 (Continued)

Channel
Description

Check

Calibrate Test

Remarks

Channel Description	Check	Calibrate	Test	Remarks
SR 3.1.1.5.1 SR 3.1.1.6.2 SR 2.1.1.6.3 10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With rod position indication 2) Log rod position indications each 4 hours when rod deviation monitor is out of service
28.i.m 11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
28.i.M 14. Boric Acid Storage Tank Level	D	R	N.A.	Note 4
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
28.i.m 16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
28.i.m 19. Boric Acid Control	N.A.	R	N.A.	
20. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
24. Accumulator Level and Pressure	S	R	N.A.	

See Chapter
3.3

SR 3.1.1.1
 SR 3.1.3.1
 SR 3.1.3.2
 SR 3.1.3.3
 SR 3.1.3.3

28.ii.i

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTSSee chapters
3.4 and 3.5

	<u>Test</u>	<u>Frequency</u>
1. Reactor Coolant Chemistry Samples	Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2. Reactor Coolant Boron	Boron Concentration	Weekly
3. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly
4. Boric Acid Storage Tank	Boron Concentration	Twice/Week⁽¹⁾
5. Control Rods SR 3.1.4.4	Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (1)
6a. Full Length Control Rod SR 3.1.4.3	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system	Monthly
6b. Full Length Control Rod SR 3.1.7.1	Move each rod through its full length to verify that the rod position indication system transitions occur	Each Refueling Shutdown
7. Pressurizer Safety Valves	Set point	Each Refueling Shutdown
8. Main Steam Safety Valves	Set point	Each Refueling Shutdown
9. Containment Isolation Trip	Functioning	Each Refueling Shutdown
10. Refueling System Interlocks	Functioning	Prior to Refueling Operations

See chapters
3.4, 3.7, 3.6, and 3.9

4.9

Reactivity Anomalies

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

36i

SR 3.1.2.2

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that

~~predicted.~~ This process of normalization should be completed after

SR 3.1.2.1

about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with predicted, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and considered an abnormal condition; its occurrence would be thoroughly investigated and evaluated.

LCO 3.1.2

The methods employed in calculating the reactivity of the core vs. burn-up and the reactivity worth of boron vs. burn-up are given in the FSAR. (1)

LCO 3.1.2

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

Reference:

(1) FSAR - Section 3.2.1

TABLE 3.5-1 (CONTINUED)
PROTECTION SYSTEM INSTRUMENTATION

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
11. Turbine Trip	3	2	2		5	50% Power
12. Deleted						
13. Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop		5	Hot Shutdown
14. Undervoltage 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	5% Power
15. Underfrequency 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	5% Power
16. Quadrant power tilt monitor (upper & lower ex-core neutron detectors)	1	NA	1		Log individual upper & lower ion chamber currents once/hr & after a load change of 10% or after 48 steps of control rod motion	Hot Shutdown

See Chapter 3.3

Amendment No. 22, 24, 41

3.5.6

LCO 3.2.4

15.49

- LCO 3.2.4
- SR 3.2.4.2
- SR 3.2.1.2
- SR 3.2.2.2
- SR 3.2.4.1
- SR 3.2.4.3

Every 24 hours verify QPTR is within limit by calculation if THERMAL POWER < 75% or verify core power distribution is acceptable by a full core flux map if THERMAL POWER ≥ 75% RTP

3.10.2 Power Distribution Limits and Misaligned Control Rod

3.10.2.1 The movable detector system shall be used to measure

SR 3.2.1.1

SR 3.2.2.1

20.XX

power distribution after each fuel reloading prior to operation of the plant at ^{75%}50% of rated power to ensure that design limits are not exceeded.

If the core is operating above 75% power with one excore nuclear channel out of service, then the quadrant

SR 3.2.1.2

SR 3.2.2.2

to average power tilt ratio shall be determined once a day by at least one of the following means:

20.XXi

- a. Movable detectors
- b. Core-exit thermocouples

Perform incore flux map to confirm hot channel factors.

3.10.2.2 Power distribution limits are expressed as hot channel

SEE CHAPTER 3.1

^{20.XXii} In MODE 1 factors. At all times, except during low power physics

LCO 3.2.1

LCO 3.2.2

tests the hot channel factors must meet the following limits in the COLR.

relocate to COLR

20.XXiii

20.XXiv

$$F_Q(Z) = (2.32/P)*K(Z) \quad \text{for } P \geq .5$$

$$F_Q(Z) = 4.64*K(Z) \quad \text{for } P \leq .5$$

$$F_{\Delta H}^N = 1.66 [1 + .3(1-P)] \quad \text{for } 0 \leq P \leq 1.00$$

Where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q shall be increased by three percent to yield F_Q . If the measured F_Q or $F_{\Delta H}^N$ exceeds the

limiting value, with due allowance for measurement

error, the maximum allowable reactor power level and

the Nuclear Overpower Trip set point shall be reduced

one percent for each percent with $F_{\Delta H}^N$ or F_Q exceeds

the limiting value, whichever is more restrictive. If

the hot channel factors cannot be reduced below the

20.XXV

20.XXvi

(20.XXV)

(72 hours)

limiting values within ~~one day~~, the Overpower ΔT trip setpoint and the Overtemperature ΔT setpoint shall be similarly reduced.

SEE CHAPTER 3.1

3.10.2.3

Except for physics tests, if the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12, then within two hours:

LCO 3.2.4

20.XXVII
20.XXVIII

- a. Correct the situation, or
- b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or
- c. Limit power to ~~75% of rated power.~~

$\geq 3\%$ below RTP for each 1% of QPTR > 1.00 .

3.10.2.4

If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

LCO 3.2.4

20.XXVII
20.XXIX

20.XXX

3.10.2.5

~~Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, within 2 hours either reduce the quadrant to average power tilt ratio to within its limit or reduce power to less than 50% of rated power. Within an additional 4 hours, either reduce the ratio to within its limit or be at hot shutdown. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip setpoint is reduced by 50%.~~

20.XXXI

3.10.2.6

Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that

SR 3.2.1.1

SR 3.2.2.1

the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by linear interpolation using the most recent measured value and the predicted value at the end of the cycle life.

20.XXXii

SR 3.2.3.3

SEE CHAPTER 3.1
3.10.2.8
LCD 3.2.3a

Except during physics tests, control rod exercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within $\pm 5\%$ of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

20.XXXiii

LCD 3.2.3
Note 1

relocated to the COLR

SEE CHAPTER 3.1
3.10.2.9
LCD 3.2.3

Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

within 15 minutes

20.XXXiv

SEE CHAPTER 3.1

3.10.2.10 Except during physics tests, control rod exercises, or excore calibration, at a power level less than or equal to 90 percent of rated power:

20.XXXV

LCO 3.2.3b

Relocate to the COLR

a.- The indicated axial flux difference may deviate from its $\pm 5\%$ target band for a maximum of one hour (cumulative) in any 24 hour period, however, the flux difference shall not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power.

LCO 3.2.3

b. If Specification 3.10.2.10a is violated, then the reactor power shall be immediately reduced to no greater than 50% power.

LCO 3.2.3

c. A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.11 A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more

LCO 3.2.3c (NOTE)

than two hours (cumulative) out of the preceding 24 hour period. One half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power.

- 3.10.2.12 When the reactor is critical and thermal power is less than or equal to 90% of rated power, an alarm is provided to indicate when the axial flux difference has been outside the target band for more than one hour (cumulative) out of any 24 hour period. In addition, when thermal power is greater than 90% of rated power, an alarm is provided to indicate when the axial flux difference is outside the target band. If either alarm is out of service, the flux difference shall be logged hourly for the first 24 hours the alarm is out of service and half-hourly thereafter.

20.XXXVI

3.10.3

Control Rod Drop Time

3.10.3.1

While critical, the individual full length (shutdown and control) rod drop time from the fully withdrawn position (indicated) shall be less than or equal to 1.8 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

- a. T_{avg} greater than or equal to 540°F, and
- b. All reactor coolant pumps operating.

3.10.3.2

With the drop time of any full length rod determined, to exceed the above limit, restore the rod drop time to within the above limit prior to criticality.

3.10.4

Control Rod Group Height

3.10.4.1

While critical, and except for physics testing, all full length (shutdown and control) rods shall be operable and positioned within ± 12 steps (indicated position) of their group step counter demand position.

SEE
CHAPTER
3.1

and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition. The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis.⁽¹⁾ In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors.

The lines shown on Figure 3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end-of-cycle life and is based on the value used in analysis of the hypothetical steam break accident. Early in cycle life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to that which is required at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

~~An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-3 has been determined from extensive analyses considering operating maneuvers consistent with the Technical Specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope~~

SEE
CHAPTER
3.1

demonstrate compliance with the Final Acceptance Criteria limit for Emergency Core Cooling Systems. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. When a measurement of $F_{\Delta H}^N$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading pattern. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which might, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these

conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 25 steps from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as described in Specification 3.10.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution limits which are given in terms of flux difference limits and control bank insertion limits are observed. Flux difference is $q_T - q_B$ as defined in Specification 2.3.1.2d.

The permitted relaxation in $F_{\Delta H}^N$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10, F_Q is arbitrarily limited for $P < 0.5$ (except for lower power physics tests).

The limits on axial power distribution described to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium

value of Axial Offset (Axial Offset = ΔI /fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies primarily with burnup. The technical specifications on power distribution assure that the F_Q upper bound envelope of 2.32 times Figure 3.10-3 is not exceeded and xenon distributions are not developed which, at a later time, could cause greater local power peaking even though the flux difference is then within the limits.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with control Bank D more than 190 steps (indicated position) withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference.

Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ± 5 percent ΔI is permitted from the indicated reference value. During periods where extensive load following is

~~required, it may be impossible to establish the required core conditions for measuring the target flux difference every month. For this reason, two methods are~~

permissible for updating the target flux difference. Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures

that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent of each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation. The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_q is depleted. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in F_Q^N and $F_{\Delta H}^N$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction on peaking factors.

The two hours in 3.10.2.3 are acceptable since complete rod misalignment (full-length control rod 12

feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If instead of determining the hot channel factors, the operator decides to reduce power, the specified 75% power maintains the design margin to core safety limits for up to 1.12 power tilt, using the 2 to 1 ratio. Reducing the overpower trip set point ensures that the protection system basis is maintained for sustained plant operation. A tilt ratio of 1.12 or more is indicative of a serious performance anomaly and a plant shutdown is prudent.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 540°F and with both reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. The various control rod banks (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the

SEE
CHAPTER
3.1

20.XX.iii

REDATE FIGURE TO THE CORE.

3.2-ii

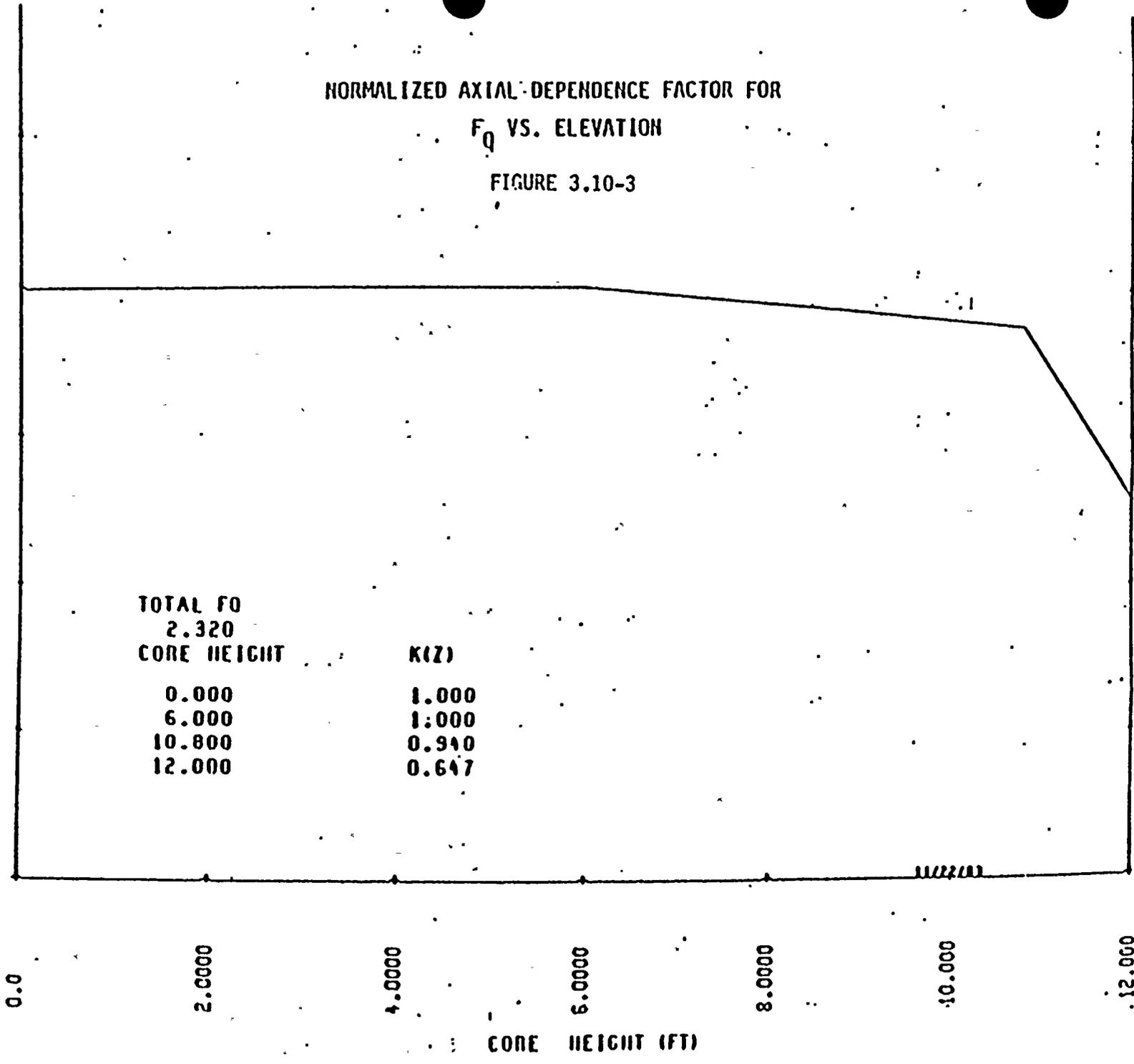
NORMALIZED AXIAL DEPENDENCE FACTOR FOR

F_Q VS. ELEVATION

FIGURE 3.10-3

1.2500
1.0000
0.7500
0.5000
0.2500
0.0

3.10-22
(Z)
K(Z)



TOTAL FO	CORE HEIGHT	K(Z)
2.320	0.000	1.000
	6.000	1.000
	10.800	0.940
	12.000	0.647

CORE HEIGHT (FT)

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS

See Chapter
3.3

<u>Channel Description</u>	<u>Check</u>	<u>Calibrate</u>	<u>Test</u>	<u>Remarks</u>
1. Nuclear Power Range	S M*(3)	D(1) Q*(3)	B/W(2)(4) P(2)(5)	1) Heat balance calculation** 2) Signal to ΔT ; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset** 4) High setpoint ($\leq 109\%$ of rated power) 5) Low setpoint ($\leq 25\%$ of rated power)
2. Nuclear Intermediate Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3. Nuclear Source Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4. Reactor Coolant Temperature	S	R	M(1) (2)	1) Overtemperature-Delta T 2) Overpower - Delta T
5. Reactor Coolant Flow	S	R	M	
6. Pressurizer Water Level	S	R	M	
7. Pressurizer Pressure	S	R	M	
8. 4 Kv Voltage & Frequency	N.A.	R	M	Reactor Protection circuits only
9. Rod Position Indication	S(1,2)	N.A.	M	1) With step counters 2) Log rod position indications each 4 hours when rod deviation monitor is out of service

* By means of the movable in-core detector system.

** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

SR 3.2.3.1 } Add new SRs
 SR 3.2.4.1 }

28.iii

3.2-17a

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

See other chapters

	<u>Test</u>	<u>Frequency</u>
1.	Reactor Coolant Chemistry Samples Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2.	Reactor Coolant Boron	Boron Concentration Weekly
3.	Refueling Water Storage Tank Water Sample	Boron Concentration Weekly
4.	Boric Acid Storage Tank	Boron Concentration Twice/Week ⁽¹⁾
5.	Control Rods	Rod drop times of all full length rods After vessel head removal and at least once per 18 months (1)
6a.	Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system Monthly
6b.	Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur Each Refueling Shutdown
7.	Pressurizer Safety Valves	Set point Each Refueling Shutdown
8.	Main Steam Safety Valves	Set point Each Refueling Shutdown
9.	Containment Isolation Trip	Functioning Each Refueling Shutdown
10.	Refueling System Interlocks	Functioning Prior to Refueling Operations

4.1

4.15

2.3 Limiting Safety System Settings,
Protective Instrumentation

Applicability:

Applies to trip settings for instruments monitoring reactor power; reactor coolant pressure, temperature, and flow; and pressurizer level.

Objective:

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:

2.3.1 Protective instrumentation for reactor trip settings shall be as follows:

LCO 3.3.1, Table 3.3.1-1

2.3.1.1 Startup Protection

FU# 2.b High flux, power range (low set point) - $\leq 25\%$ of rated power.

2.3.1.2 Core Protection

FU# 2.a a. High flux, power range (high set point) - $\leq 109\%$ of rated power.

FU# 7.b b. High pressurizer pressure - ≤ 2385 psig.

FU# 7.a c. Low pressurizer pressure - ≥ 1865 psig.

LCO 3.3.1

FU# 5 and
NOTE 1

d. Overtemperature ΔT

$$\Delta T_o \cdot [K_1 + K_2 (P - P^1) - K_3 (T - T^1) \left(\frac{1 + \tau_1 S}{1 + \tau_2 S} \right)] - f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = 573.5°F

P = pressurizer pressure, psig

P^1 = 2235 psig

K_1 = 1.20

K_2 = .000900

K_3 = .0209

τ_1 = 25 sec

τ_2 = 5 sec

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ less than +13 percent, $f(\Delta I) = 0$

Note (i)
to Table 2
3.3.1-1

(ii) for each percent that the magnitude of $q_t - q_p$ is more positive than +13 percent, the ΔT trip set point shall be automatically reduced by equivalent of 1.3 percent of rated power.

Note (1) to Table 3.3.1-1

FU #6

e. Overpower ΔT

$$\Delta T_o [K_4 - K_5 (T - T^1) - K_6 \frac{\tau_3 ST}{\tau_3 S + 1}] - f(\Delta I)$$

where

- ΔT_o = indicated ΔT at rated power, °F
- T = average temperature, °F
- T^1 = indicated T avg at nominal conditions at rated power, °F
- K_4 = 1.077
- K_5 = 0.0 for $T < T^1$
= 0.0011 for $T \geq T^1$
- K_6 = 0.0262 for increasing T
= 0.0 for decreasing T
- τ_3 = 10 sec
- $f(\Delta I)$ = as defined in 2.3.1.2.d

Note (2) to Table 3.3.1-1

LCO 3.3.1

FU# 9

- f. Low reactor coolant flow - $\geq 90\%$ of normal indicated flow.
- g. Low reactor coolant pump frequency - ≥ 57.5 Hz.

2.3.1.3 Other reactor trips

FU# 8

- a. High pressurizer water level - $\leq 88\%$ of span.

FU# 13

- b. Low-low steam generator water level - $\geq 6\%$ of narrow range instrument span

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

- 2.3.2.1 Remove bypass of "at power" reactor trips at high power (low pressurizer pressure and low reactor coolant flow) for both loops:

FU# 16.b

- Power range nuclear flux - $\leq 8.5\%$ of rated power
(1) (Note: During cold rod drop tests; the pressurizer high level trip may be bypassed.)

- 2.3.2.2 Remove bypass of single loss of flow trip at high power:

FU# 16.c

- Power range nuclear flux - $\leq 50\%$ rated power
Relay operating will be tested to insure that they perform according to their design characteristics which must fall in within the ranges defined below:

- 2.3.3.1 Loss of voltage relay operating time ≤ 8.5 seconds for 480 volt safeguards bus voltages ≤ 368 volts.

SR 3.3.4.2

Measured values shall fall at least 5% below the theoretical limit. This 5% margin is shown as the 5% tolerance curve in Figure 2.3-1.

4.iv

2.3.3:2 Acceptable degraded voltage relay operating times and setpoints, for 480 volt safeguards bus voltages ≤ 414 volts and > 368 volts are defined by the safeguard

equipment thermal capability curve shown in Figure

2.3-1. Measured values shall fall at least 5% below

the theoretical limit. This 5% margin is shown as the 5% tolerance curve in Figure 2.3-1.

4.iv

Basis:

The high flux reactor trip (low set point) provides redundant protection in the power range for a power excursion beginning from low power. This trip value was used in the safety analysis. (1) In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. (3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that:

(1) the transient is slow with respect to the thermal capacity of the reactor coolant system to respond to power increases (1)(2) and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors, (2) is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced. (4)

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur as described in Section 7.2 of the UFSAR. This setpoint includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors. (1)

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or both reactor coolant pumps. The set points specified are consistent with the value used in the accident analysis. (1)

The underfrequency reactor trip protects against a decrease in flow caused by low electrical frequency. The specified set point assures a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 700 ft.³ of water corresponds to 92% of span. A trip at this set point contains margin for both normal instrument error and transient overshoot of level beyond this trip setting. An additional 4% instrument error has been assumed to account for the effects of elevated temperatures on level measurement in accordance with IE Bulletin 79-21.⁽¹²⁾ Therefore a trip setpoint of 88% prevents the water level from reaching the safety valves.⁽⁴⁾

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. A set point of 5% is equivalent to at least 40,000 lbs. of water and assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.⁽⁵⁾ An additional 11% has been added to the set point to account for error which may be introduced into the steam generator level system at a containment temperature of 286°F as determined by evaluation performed for temperature effects on level measurements required by IE Bulletin 79-21.

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

Operation with one pump will not be permitted above 130 MWT (8.5%). An orderly power reduction to less than 130 MWT (8.5%) will be accomplished if a pump is lost while operating between 130 MWT (8.5%) and 50%. Automatic protection is provided so that a power-to-flow ratio is maintained equal to or less than one, which insures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase. For this reason the single pump loss of flow trip can be bypassed below 50% power.

The loss of voltage and degraded voltage trips ensure operability of safeguards equipment during a postulated design basis event concurrent with a degraded bus voltage condition. (9)(10)(11)

The undervoltage set points have been selected so that safeguards motors will start and accelerate the driven loads (pumps) within the required time and will be able to perform for long periods of time at degraded conditions above the trip set points without significant loss of design life. All control circuitry or safety related control centers and load centers, except for motor control centers M and L, are d.c. Therefore, degraded grid voltages do not affect these control centers and load centers. Motor control centers M and L, which supply the Standby Auxiliary Feedwater System, are fully protected by the undervoltage set points. Further, the Standby System is normally not in service and is manually operated only in total loss of feedwater and auxiliary feedwater.

The 5% tolerance curve in Figure 2.3-1 and the requirements of specifications 2.3.3.1 and 2.3.3.2 include 5% allowance for measurement error. Thus, providing the measurement error is less than 5%, measured values may be directly compared to the curve. If measurement error exceeds 5%, appropriate allowance shall be made.

~~2.3-8c~~

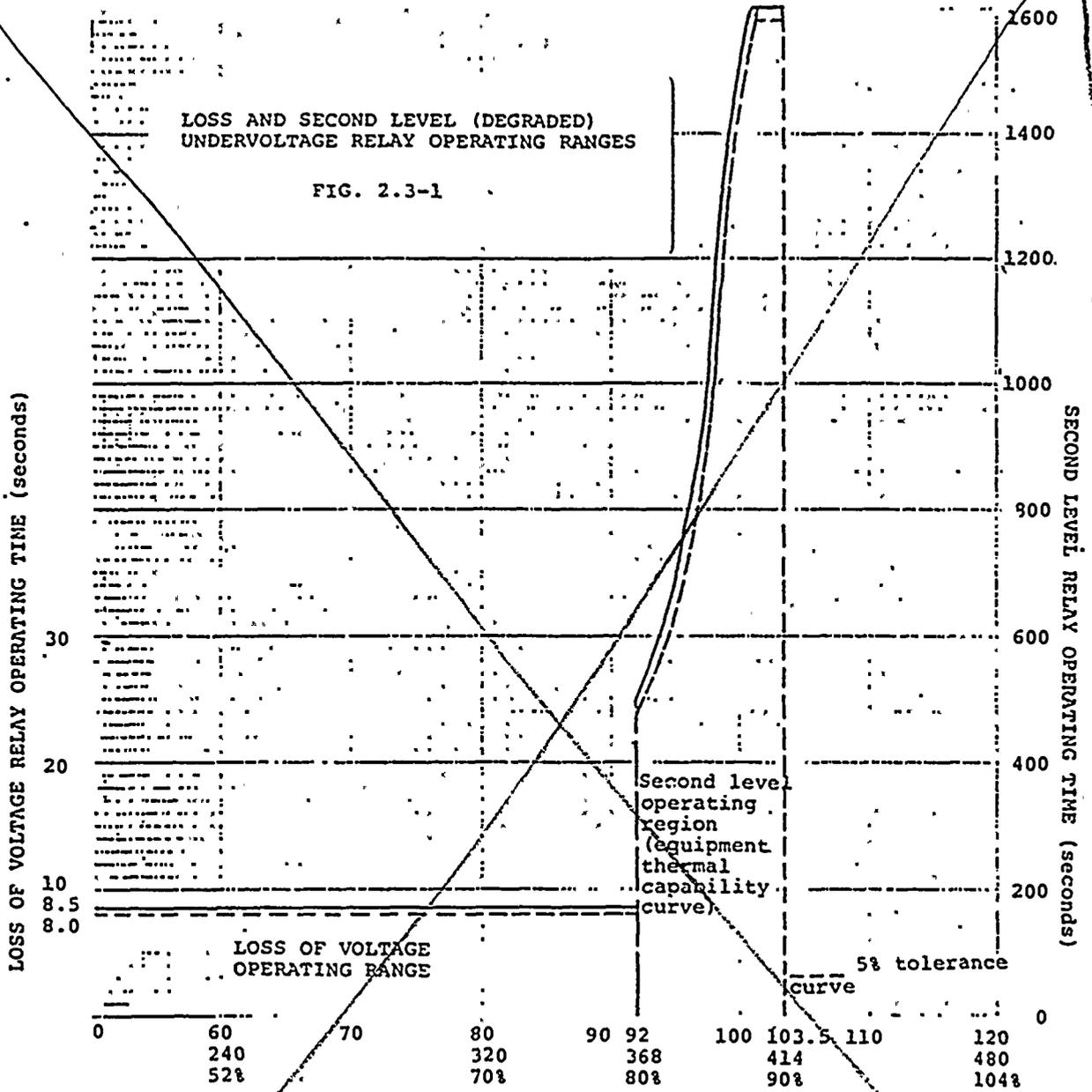
References:

- (1) UFSAR 15.0
- (2) UFSAR 15.4
- (3) UFSAR 15.6
- (4) UFSAR 7.2
- (5) UFSAR 15.2
- (6) Deleted
- (7) Deleted
- (8) Deleted
- (9) Letter from L.D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977.
- (10) Letter from L.D. White, Jr. to A. Schwencer, NRC, dated September 30, 1977.
- (11) Letter from L.D. White, Jr. to D. Ziemann, NRC, dated July 24, 1978.
- (12) Letter from L.D. White, Jr. to B. Grier, USNRC dated September 14, 1979.

4.1v

LOSS AND SECOND LEVEL (DEGRADED) UNDERVOLTAGE RELAY OPERATING RANGES

FIG. 2.3-1



LOSS OF VOLTAGE RELAY OPERATING TIME (seconds)

SECOND LEVEL RELAY OPERATING TIME (seconds)

Secondary Volts (120V)
Primary Volts (480)
1 Volts (460" Base)

SAFEGUARDS BUS VOLTAGE

3.5 Instrumentation Systems

Objective

To delineate the conditions of the plant instrumentation and safety circuits.

Specification

- LCO 3.3.1 3.5.1 Protection System Instrumentation
- 3.5.1.1 The Protection System Instrumentation shown on Table 3.5-1 shall be operable whenever the conditions specified in Column 6 are exceeded.
- LCO 3.3.1 3.5.1.2 In the event the number of channels of a particular sub-system falls below the limits given in the columns 1 or 3 of Table 3.5-1, action shall be taken according to the requirements shown in column 5 of Table 3.5-1.
(15.i.a)
- 3.5.2 Engineered Safety Feature Actuation Instrumentation
- LCO 3.3.2 3.5.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Table 3.5-2 shall be operable with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5-4, whenever the conditions specified in column 6 of Table 3.5-2 are exceeded.
- LCO 3.3.2 3.5.2.2 In the event the number of channels of a particular subsystem falls below the limits given in columns 1 or 3 of Table 3.5-2, action shall be taken according to the requirements of column 5 of Table 3.5-2.
(15.ii.a)
- 3.5.2.3 ~~With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5-4, declare the channel inoperable and take action according to the requirements of column 5 of Table 3.5-2 until the channel is restored to operable status with the trip setpoint adjusted consistent with the Trip Setpoint value.~~
(15.ii.b)
- LCO 3.3.3 3.5.3 Accident Monitoring Instrumentation
- 3.5.3.1 The accident monitoring instrumentation channels shown in Table 3.5-3 shall be operable whenever the reactor is at or above hot shutdown.



3.5.3.2
 LCO 3.3.3
 Cond A
 Cond C (15.iii.a)

When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5-3, either restore the inoperable channel(s) to operable status within 30 days (15.iii.b), or be in at least hot shutdown within the next 12 hours. (15.iii.c) Cond C

Cond B
 Cond C/G (for Functions with only one channel)

3.5.3.3
 LCO 3.3.3
 Cond D (15.iii.a)

When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table 3.5-3 either restore the inoperable channel(s) to operable status within 7 days (15.iii.d) or be in at least hot shutdown within the next 12 hours. (one inoperable channel)

Cond G

3.5.4 (15.iv)

The radiation accident monitoring instrumentation channels shown in Table 3.5-6 shall be operable, whenever the reactor is at or above hot shutdown. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.5-6. Startup may commence or continue consistent with the action statement.

3.5.5 Radioactive Effluent Monitoring Instrumentation

3.5.5.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-5 shall be operable at all times with alarm and/or trip setpoints set to insure that the limits of Specification 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

For R-11 and R-12 only. All other instrumentation addressed in 5.0

For R-11 and R-12 only.
All others addressed in
5.0

3.5.5.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

- (i) the setpoint shall be immediately corrected without declaring the channels inoperable; or
- (ii) immediately suspend the release of effluents monitored by the effected channel; or
- (iii) declare the channel inoperable.

3.5.5.3 If the number of channels which are operable is found to be less than required, take the action shown in Table

3.5-5. Exert best efforts to return the instruments to OPERABLE status within 31 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

Reverted
to 3.3.4

3.5.6 Control Room HVAC Detection Systems

3.5.6.1 During all modes of plant operation, detection systems

for ~~chlorine gas, ammonia gas and~~ radioactivity in the

15.V

control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

Table 3.3.5-1

(15.v)

chlorine, < 5 ppm
ammonia, < 35 mg/m³radioactivity, particulate $\leq 1 \times 10^{-8}$ μ Ci/cciodine $\leq 9 \times 10^{-9}$ μ Ci/ccnoble gas $\leq 1 \times 10^{-5}$ μ Ci/cc

(15.vii)

3.5.6.2

LCO 3.3.5

Cond A and

R.F. Note

With one of the detection systems inoperable, within 1 hour isolate the control room HVAC air intake. Maintain the air intake isolated except for short periods, not to exceed 1 hour a day, when fresh air makeup is allowed to improve the working environment in the control room.

Add Cond B
Cond C

(15.vi)

Basis

During plant operations, the complete instrumentation system will normally be operable. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels inoperable since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is inoperable.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50.

Control Room HVAC detection systems are designed to prevent the intake of chlorine, ammonia and radiation at concentrations which may prevent plant operators from performing their required functions. Concentrations which initiate isolation of the control room HVAC system have been established using the guidance of several established references (2-4).

The chlorine isolation setpoint is 1/3 of the toxicity limit of reference 2 but slightly greater than the short term exposure limit of reference 4. The ammonia setpoint is established at approximately 1/3 of the toxicity limit for anhydrous ammonia in reference 2 and equal to the short term exposure limit of reference 4.

The setpoints for radioactivity correspond to the

6.5

maximum permissible concentrations of reference 3 for Cs-137, I-131 and Kr-85.

The mini-purge system is connected to the plant vent. 10 CFR Part 100 type releases via mini-purge are limited by an isolation signal generated from SI. 10 CFR Part 20 releases from mini-purge are considered to be similar to other plant ventilation releases and are monitored by R-10B, R-13, and R-14. R-14A may be a substitute for R-10B. Automatic isolation of mini-purge for 10 CFR Part 20 type releases is considered unnecessary due to the low flow associated with mini-purge and the continuous monitoring. However, the automatic isolation provisions using R-11 or R-12 provide additional margin for 10 CFR Part 20 type releases. Therefore, R-11 or R-12 is required to sample containment during mini-purge operation. To ensure the containment sample monitored by R-11 or R-12 is representative of the containment atmosphere, at least one recirculation fan is required to be in operation during mini-purge operation. Should R-11 and/or R-12 become inoperable, a 1 hour limit is chosen to be consistent with the generally accepted time for prompt action.

3.5-4a

References

- 1) Updated FSAR - Section 7.2.
- 2) USNRC Regulatory Guide 1.78, June 1974, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.
- 3) 10 CFR 20 Appendix B, Table I.
- 4) Threshold Limit Values for Chemical Substances and Physical Agents in the Work Environment, 1982. Published by American Conference of Governmental Industrial Hygienists.

~~3-5-4b~~

INSTRUMENT NO. 7

LC03.3.1
TABLE 3.3.1-1

TABLE 3.5-1
PROTECTION SYSTEM INSTRUMENTATION

		1	2	3	4	5	6	
		TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE	
FU #1	1. Manual	2	1	2		1		
FU #2	2. Nuclear Flux Power Range	4	2	3	<p>For low setting, 2 of 4 power range channels greater than 10% F.P. (15.i.h)</p> <p>Fu #2.a high setting (15.i.j)</p> <p>2 of 4 power range channels greater than 10% F.P. (15.i.j)</p> <p>1 of 2 intermediate range channels greater than 10⁻¹⁰ amps. (15.i.k)</p>	2 Note 1	when RCCA is withdrawn (15.i.b)	
	Fu #2.b low setting	4	2	3			2	when RCCA is withdrawn (15.i.d)
	Fu #2.a high setting	4	2	3			2	when RCCA is withdrawn (15.i.j)
FU #3	3. Nuclear Flux Intermediate Range	2	1	1		3 Note 1	when RCCA is withdrawn (15.i.j)	
FU #4	4. Nuclear Flux Source Range	2	1	2		4 Note 1	Note 2 (15.i.k)	
		2	0	1		4	Note 3	
FU #5	5. Overtemperature Δ T	4	2	3		2	Hot Shutdown	
FU #6	6. Overpower Δ T	4	2	3		2	Hot Shutdown	
FU #7.a	7. Low Pressurizer Pressure	4	2	3		2	8.5% power (15.i.i)	
FU #7.b	8. Hi Pressurizer Pressure	3	2	2		5	Hot Shutdown	
FU #8	9. Pressurizer-Hi Water Level	3	2	2		5	5% power (15.i.i)	
FU #9.a	10. Low Flow in one loop (> 50% F.P.)	3/loop	2/loop	2/loop		5	5% power 8.5%	
FU #9.b	Low Flow both loops (8.5% - 50% F.P.)	3/loop	2/loop	2/loop		6	5% power 8.5%	

TABLE 3.5-1 (CONTINUED)
PROTECTION SYSTEM INSTRUMENTATION

	1	2	3	4	5	6
	TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE
15.i.a						
15.i.c						
NO. FUNCTIONAL UNIT						
FU#11 Turbine Trip (Low Pressure Stop Pressure)	3	2	2		5	50% Power
12 Deleted						
FU#13 Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop		5	Hot Shutdown
FU#14 Undervoltage 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	8.5% 5% Power
15. Underfrequency 4 KV Bus	2/bus	1/bus (both busses)	2/bus (on either bus)		6	8.5% 5% Power

Required c. channel

15.i.b

15.i.i

16. Quadrant power tilt monitor (upper & lower ex-core neutron detectors)

1 NA 1

Log individual upper & lower ion chamber currents once/hr & after a load change of 10% or after 48 steps of control rod motion

Hot Shutdown

Addressed with Chapter 3.2

Add Function # 10, "RCP Breaker Position" 15.i.w

Add Function # 15, "SI Input from ESFAS" 15.i.x

Amendment No. 24

TABLE 3.5-1 (Continued)
PROTECTION SYSTEM INSTRUMENTATION

NO. FUNCTIONAL UNIT	1	2	3	4	5	6
	TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE
17. Circulating Water Flood Protection						
a. Condenser	2 sets of 3	2 of 3 in either set	2 of 3 in both sets		Power operation may be continued for a period of up to 7 days with 1 channel (1 set of three) inoperable or for a period of 24 hrs. with two channels (2 sets of three) inoperable. Otherwise be in hot shutdown in an additional 6 hours.	Hot Shutdown
(15.i.g)						
b. Screenhouse	2 sets of 3	2 of 3 in either set	2 of 3 in both sets		Power operation may be continued for a period of up to 7 days with 1 channel (1 set of three) inoperable or for a period of 24 hrs. with two channels (2 sets of three) inoperable. Otherwise be in hot shutdown in an additional 6 hours.	Hot Shutdown
LC03.3.4						
18. Loss of Voltage 480V Safeguards Bus	2 sets of 2/bus	1 of 2 in each set in one bus	2 of 2 in one of the two sets		7	T _{RCS} = 350°F
(15.i.v)						
(15.i.a)						(15.i.f)

TABLE 3.5-1 (Continued)
PROTECTION SYSTEM INSTRUMENTATION

		1	2	3	4	5	6
	NO. FUNCTIONAL UNIT	TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE
LC03.3.4	19. Degraded Voltage 480V Safeguards Bus	2/bus	2/bus	1/bus		7	T _{max} = 350°F
FU# 17 FU# 18 FU# 19	20. Automatic Trip Logic Including Reactor Trip Breakers	2	1	2	Note 4	14	Note 5

NOTE 1: When block condition exists, maintain normal operation.

FU#4
Notes (a) & (c) NOTE 2: Channels should be operable at all modes below the bypass condition with the reactor trip system breakers in the closed position and control rod drive system capable of rod withdrawal.

FU#4
Note (e) NOTE 3: Channels shall be operable at all modes below the bypass condition except during refueling defined to be when fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

FU#17
Con. Note 1 NOTE 4: One reactor trip breaker may be bypassed for surveillance testing provided the other reactor trip breaker is operable.

FU#17
"18 (a)
"19 NOTE 5: Channels shall be operable at all modes above refueling when the control rod drive system is capable of rod withdrawal unless both reactor trip breakers are open.

F.P. = Full Power

TABLE 3.5-2
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

Attachment No. 22

LCO 3.3.2
Table 3.3.2-1

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
FU # 1.a 1. SAFETY INJECTION a. Manual	2	1	2		8	$T_{RCS} = 350^{\circ}F$
FU # 1.c b. High Containment Pressure	3	2	2		9	$T_{RCS} = 200^{\circ}F$ $350^{\circ}F$
FU # 1.e c. Steam Generator Low Steam Pressure/Loop	3	2	2	Primary pressure less than 2000 psig	9	$T_{RCS} = 350^{\circ}F$
FU # 1.d d. Pressurizer Low Pressure	3	2	2	Primary pressure less than 2000 psig	9	$T_{RCS} = 350^{\circ}F$
FU # 2.a 2. CONTAINMENT SPRAY a. Manual	2	2**	2		10	Cold Shutdown
FU # 2.c b. Hi-Hi Containment Pressure (Containment Spray)	2 sets of 3	2 of 3 in both sets	2 per set in either set		11	Cold Shutdown

15.ii.a

Required channels

SR 3.3.2.6
15.ii.d
LCO 3.3.2 Footnote (a)

15.ii.b

15.ii.c

** Must actuate 2 switches simultaneously. 15.ii.a

Add Function # 1.b, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

Add Function # 2.b, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

TABLE 3.5-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
<p>2 #6.a</p> <p>3. AUXILIARY FEEDWATER Motor and Turbine Driven</p> <p>a. Manual</p>	1/pump	1/pump	1/pump	15.ii.k	<p>Cond N</p> <p>Declare pump inoperable per LEU 3.2.5</p>	T _{RCS} = 350°F
<p>FU#6.c</p> <p>b. Stm. Gen. Water Level-low-low</p> <p>i. Start Motor Driven Pumps</p>	3/stm.gen.	2/stm.gen. either gen.	2/stm.gen. both gen.	15.ii.o	12	T _{RCS} = 350°F
<p>FU#6.d</p> <p>ii. Start Turbine Driven Pump</p>	3/stm.gen.	2/stm.gen. both gen.	2/stm.gen. either gen.	12	12	T _{RCS} = 350°F
<p>FU#6.d</p> <p>c. Loss of 4 KV Voltage Start Turbine Driven Pump</p>	2/bus	1/bus (both buses)	2/bus (either bus)	12	12	T _{RCS} = 350°F
<p>FU#6.c</p> <p>d. Safety Injection Start Motor Driven Pumps</p>	(see Item 1)					
<p>FU#6.e</p> <p>e. Trip of both Feed- water Pumps starts Motor Driven Pumps</p>	2/pump	1/pump both pumps	2/pump either pump		15.ii.n	5% power
<p>2 #6.a</p> <p>Standby Motor Driven</p> <p>a. Manual</p>	1/pump	1/pump	1/pump	15.ii.k	<p>Cond N</p> <p>Declare pumps inoperable per LEU 3.2.5</p>	T _{RCS} = 350°F

Add Function #6.b, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

REFERENCE NO. 22

TABLE 3.5-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO. FUNCTIONAL UNIT	1 TOTAL NO. of CHANNELS	2 NO. of CHANNELS TO TRIP*	3 MIN. OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	6 CHANNEL OPERABLE ABOVE
4. CONTAINMENT ISOLATION						
4.1 Containment Isolation						
FU#3,a a. Manual	2	1	2		10	Cold Shutdown
FU#3,c b. Safety Injection (Auto Actuation)		(See Table 3.5-2, Item 1)				
Table 4.2 3.3.5-1 Containment Ventilation Isolation						
a. Manual	2	1	1		13	Cold Shutdown
b. High Containment Radioactivity	2	1	2		13	Cold Shutdown
c. Manual Spray		(See Table 3.5-2, Item 2a)				
d. Safety Injection		(See Table 3.5-2, Item 1)				

Requires Channels

15.ii.a

15.ii.b

15.ii.p

Add Function # 3, b, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

TABLE 3.5-2 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION INSTRUMENTATION

NO. FUNCTIONAL UNIT	1	2	3	4	5	6
	TOTAL NO. of CHANNELS	NO. of CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 3 CANNOT BE MET	CHANNEL OPERABLE ABOVE
5. STEAM LINE ISOLATION						
FU#4.e a. Hi-Hi Steam Flow with Safety Injection Footnote (c)	2 Hi-Hi SF with S.I. for each loop	1 SF with S.I. in each loop	***		12	*T _{RCS} = 350°F w/MSIV's open
FU#4.d b. Hi Steam Flow and 2 of 4 Low T _{AVG} with Safety Injection Footnote (c)	2 Hi SF and 4 Low T _{AVG} with S.I. for each loop	1 Hi SF and 2 Low T _{AVG} with S.I. for each loop	***		12	*T _{RCS} = 350°F w/MSIV's open
FU#4.c c. Containment Pressure Footnote (c)	3	2	2		9	*T _{RCS} = 350°F w/MSIV's open
FU#4.a d. Manual Footnote (c)	1/loop	1/loop	1/loop		8	*T _{RCS} = 350°F w/MSIV's open
6. FEEDWATER LINE ISOLATION						
FU#5.c a. Safety Injection		(See Table 3.5-2, Item 1)				
FU#5.b b. Hi Steam Generator Level Footnote (d)	3/loop	2/loop in either loop	2/loop in both loops		9	***T _{RCS} = 350°F w/FW Isol valves open

Proposed Channels

15.ii.a

15.ii.b

Note (c) * RCS temperature may be above 350°F if MSIV's are closed.
Note (d) *** RCS temperature may be above 350°F if FW Isol. valves are closed.

*** Both trains must be capable of providing a S.I. signal to each loop. 15.ii.a

Add Function 4.b, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

Add Function 5.a, "Automatic Actuation Logic and Actuation Relays" 15.ii.g

Add Function 7, "ESFAS Pressurizer Pressure Interlock" 15.ii.d

3-2-81

ACTION STATEMENTS

LCO 3.3.1

Cond B
Cond C (15.i.d)

1. With the number of operable channels one less than the Minimum Operable Channels requirement, restore the inoperable channel to operable status within 48 hours or be in hot shutdown with all RCCA's fully inserted within the next 6 hours.

LCO 3.3.1

Cond D
Cond K
Cond M

2. With the number of operable channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 2 hour and the requirements for the minimum number of channels operable are satisfied. However, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels.

(6)
(15.i.f)

(4) (15.i.g)

Cond F

With the number of operable channels less than the Minimum Operable Channels requirement, be at a condition where operability is not required according to Column 6 of Table 3.5-1 within 6 hours.

either reduce THERMAL POWER to < 5E+11watts or increase power to > 87% RTP in 2 hours

LCO 3.3.1

Cond E/F

3. With the number of operable channels one less than the Minimum Operable Channels requirement, suspend all operations involving positive reactivity changes and have all RCCA's fully inserted within 6 hours.

(15.i.j)

LCO 3.3.1

Cond G/H/I/J

4. With the number of operable channels one less than the Minimum Operable Channels requirement, suspend all operations involving positive reactivity changes. If the channel is not restored to operable status within 48 hours, open the reactor trip breaker within the next hour.

(15.i.k)

ADD RAL.2

(6) (15.l.2)

LCO 3.3.1

Cond D
Cond P
Cond Q
Cond F

5. With the number of operable channels one less than the Total Number of Channels, operation may proceed until the next Channel Functional Test provided the inoperable channel is placed in the tripped condition within 2 hour. With the number of operable channels one less than the Minimum Operable Channels requirement, or at the time of the next required Channel Functional Test referenced above, be at a condition where channel operability is not required according to Column 6 of Table 3.5-1 within the next 6 hours.

(15.i.m)

(15.i.m)

← Add Note (15.i.m)

LCO 3.3.1

Cond M, O, K, L

6. With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 2 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of a reactor trip signal, operation may proceed until this Channel Functional Test. At the time of this next Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels, be at a condition where channel operability is not required according to Column 6 of Table 3.5-1 within the next 6 hours.

(15.i.n) (6)

(15.l.o)

← Add Note (15.i.o)

LCO 3.3.4 7.
Cond A

With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 6 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of a trip signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels, either

SR Note 15.i.t

15.i.u

Cond B
Cond C

- a) be at Hot Shutdown within the next 6 hours and an RCS temperature less than 350°F within the following 6 hours, or
- b) energize the affected bus with a diesel generator.

LCO 3.3.2 8.
Cond G
Cond D
Cond B
Cond C

With the number of operable channels one less than the Minimum Operable Channels required, restore the inoperable channel to operable status within 48 hours or be in Hot Shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

LCO 3.3.2 9.
Cond L
Cond M
Cond N
Cond O
Cond P
Cond Q
Cond R
Cond S
Cond T
Cond U
Cond V
Cond W
Cond X
Cond Y
Cond Z

With the number of operable channels one less than the Total Number of Channels required, operation may proceed until the next Channel Functional Test provided the inoperable channel is placed in the tripped position within 6 hour. At the next Channel Functional Test, or at any time the number of operable channels is less than the Minimum Operable Channels required, be at Hot Shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

15.ii.c
Note

(30 hours to be in MODES for 2 - Containment Pressure - High) 15.ii.c

LCO 3.3.2 10.
Cond H
Cond K

With the number of operable channels one less than the Minimum Operable Channels required, restore the inoperable channel to operable status within 48 hours or be in Hot Shutdown within an additional 6 hours, and at cold shutdown within the following 30 hours.

Note

15.ii.i

72 15.ii.i

LCO 3.3.2 11.
Cond I
Cond L

With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 2 hours. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of an actuation signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels required, be at Hot Shutdown within 6 hours and at Cold Shutdown within the following 30 hours.

15.ii.b

48

15.ii.m

LCO 3.3.2 12.
Cond F
Cond G

With the number of operable channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. Should the next Channel Functional Test require the bypass of an inoperable channel to avoid the generation of an actuation signal, operation may proceed until this Channel Functional Test. At the time of this Channel Functional Test, or if at any time the number of operable channels is less than the Minimum Operable Channels required, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F within 6 hours.

NOTE 15.ii.l
Cond D
Cond K

LCO 3.3.5
Cond A
Cond B
Cond C

13. With the number of operable channels less than the Minimum Operable Channels required, operation may continue provided the containment purge and exhaust valves are maintained closed: within 4 hours ~ 15.ii.x

15.i.aa & 15.i.cc

15.i.z & 15.i.bb

LCO 3.3.1 14.
Cond T Cond R
Cond V

Should one reactor trip breaker or channel of trip logic be inoperable the plant must not be in the operating mode following a six hour time period, and the breaker must be open.

LCO 3.3.1
Cond W
Cond V
Cond X
Cond A

If one of the diverse reactor trip breaker trip features (undervoltage or shunt trip attachment) on one breaker is inoperable, restore it to operable status within 48 hours or declare breaker inoperable. If at the end of the 48 hour period one trip feature is inoperable it must be repaired or the plant must not be in the operating mode, and the reactor trip breaker must be open, following an additional six hour time period. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to operable status.

15.ii.dd

Cond T, Note 2

15.i.ee

Replace w/ LCO 3.3.3,
Table 3.3.3-1
Functions #1-#22

15.iii.e

15.iii.a

TABLE 3.5-3
Accident Monitoring Instrumentation

	TOTAL REQUIRED NO. OF CHANNELS (7)	MINIMUM CHANNELS OPERABLE (7)
1. Pressurizer Water Level (1)	2	1
2. Auxiliary Feedwater Flow Rate (2)(3)	2/steam generator	1/steam generator
3. Steam Generator Water Level - Wide Range (3)	1/steam generator	1/steam generator
4. Reactor Coolant System Subcooling-Margin Monitor (4)	2	1
5. Pressurizer PORV Position Indicator (5)	2/Valve	1/Valve
6. PORV Block Valve Position Indicator (1)	1/Valve	0/Valve
7. Pressurizer Safety Valve Position Indicator (5)	2/Valve	1/Valve
8. Containment Pressure (8)	2	1
9. Containment Water Level (Narrow Range, Sump A)	1(6)	1(6)
10. Containment Water Level (Wide Range, Sump B)	2	1
11. Core-Exit Thermocouples	4/core quadrant	2/core quadrant
12. Reactor Vessel Level Indication System	2	1

Notes

- (1) Emergency power for pressurizer equipment, NUREG-0737, item II.G.1.
- (2) Auxiliary feedwater system flow indication, NUREG-0737, item II.E.1.2.
- (3) Only 2 out of the 3 indications (two steam generator auxiliary feedwater flow and one wide-range steam generator level) are required to be operable, NUREG-0737, item II.E.1.2.
- (4) Instrumentation for detection of inadequate core cooling, NUREG-0737, item II.F.2.1.
- (5) Direct indication of relief and safety valve position, NUREG-0737, item II.D.3. Two channels include a primary detector and RTD as the backup detector.
- (6) Operation may continue with less than the minimum channels operable provided that the requirements of Technical Specification 3.1.5.1 are met.
- (7) See Specification 3.5.3 for required action.
- (8) Containment pressure monitor, NUREG-0737, item II.F.1.4.



TABLE 3.5-4
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Amendment No. 1

	<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES*</u>
LC03.3.2	1. SAFETY INJECTION AND FEEDWATER ISOLATION		
FU# 1.a	a. Manual Initiation	Not Applicable	Not Applicable
FU# 1.c	b. High Containment Pressure	≤ 4.0 psig	6.0 ≤ 5.0 psig
FU# 1.d	c. Low Pressurizer Pressure	≥ ¹⁷⁵⁰ 1723 psig	≥ 1715 psig
FU# 1.e	d. Low Steam Line Pressure	≥ 514 psig	358 ≥ 514 psig
LC03.3.2	2. CONTAINMENT SPRAY		
FU# 2.a	a. Manual Initiation	Not Applicable	Not Applicable
FU# 2.c	b. High-High Containment Pressure	≤ 28 psig	32.5 ≤ 28 psig
LC03.3.2	3. CONTAINMENT ISOLATION		
	a. Containment Isolation		
FU# 3.a	1. Manual	Not Applicable	Not Applicable
FU# 3.c	2. From Safety Injection Automatic Actuation Logic	Not Applicable	Not Applicable
	b. Containment Ventilation Isolation		
LC03.3.5	1. Manual	Not Applicable	Not Applicable
15.ii.p	2. High Containment Radioactivity	Note 3	Not Applicable
	3. From Safety Injection	Not Applicable	Not Applicable
	4. Manual Spray	Not Applicable	Not Applicable

Amendment No. 23

TABLE 3.5-4 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES*</u>
LCO 3.3.2 4. STEAM LINE ISOLATION		
FU# 4.a a. Manual	Not Applicable	Not Applicable
FU# 4.c b. High Containment Pressure	≤ 18 psig	≤ 20 psig
FU# 4.d c. High Steam Flow, Coincident with Low T_{avg} and SI	dp corresponding to $\leq 0.40 \times 10^6$ lbs/hr at 755 psig $T_{avg} \geq 545^\circ\text{F}$	dp corresponding to $\leq 0.55 \times 10^6$ lbs/hr at 755 psig $T_{avg} \geq 543^\circ\text{F}$
FU# 4.e d. High-High Steam Line Flow Coincident with SI	dp corresponding $\leq 3.6 \times 10^6$ lbs/hr at 755 psig	dp corresponding to $\leq 3.7 \times 10^6$ lbs/hr at 755 psig
LCO 3.3.2 5. FEED WATER ISOLATION		
FU# 5.b a. High Steam Generator Water Level	$\leq 67\%$ of narrow range instrument span each steam generator	$\leq 68\%$ of narrow range instrument span each steam generator
LCO 3.3.2 6. AUXILIARY FEEDWATER		
FU# 6.b a. Low-Low Steam Generator Water Level	$\geq 17\%$ of narrow range instrument span each steam generator	$\geq 16\%$ of narrow range instrument span each steam generator. See Note 1.
FU# 6.c b. From Safety Injection	N.A.	N.A.
FU# 6.d c. Loss of 4 kV Voltage (Start TAFP)	62% of 4160 volts Note 2	Note 2
FU# 6.e d. Feedwater Pump Breakers Open (start MAFP)	Not Applicable	Not Applicable

α
β
γ
δ

TABLE 3.5-4 (Continued)
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
LC03.3.4 7. LOSS OF VOLTAGE		
SR 3.3.4.2 a. 480 V Safeguards Bus Under-voltage (Loss of Voltage)	see Figure 2.3-1	15. ii. c
SR 3.3.4.2 b. 480 V Safeguards Bus Under-voltage (Degraded Voltage)	see Figure 2.3-1	
LC03.3.2 8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
SR 3.3.2.6 a. Pressurizer Pressure, (block, unblock SI)	≤2000 psig	≤2000 psig

15. ii. s

Note 1: A positive 11% error has been included in the setpoint to account for errors which may be introduced into the steam generator level measurement system at a containment temperature of 286°F as determined by an evaluation performed on temperature effects on level systems as required by IE Bulletin 79-21.

15. ii. s

Note 2: This setpoint value is from inverse time curve for OVT relay (406C883) with tap setting of 82 volts and time dial setting of 1. Delay at 62% voltage is 3.6 seconds. The allowable values are ±5% of the trip setpoint.

Note 3: The trip setpoints for containment ventilation isolation while purging shall be established to correspond to the limits of 10 CFR Part 20 for unrestricted areas. The setpoints are determined procedurally in accordance with Technical Specification 3.9.2 by calculating effluent monitor count rate limits, which take into account appropriate factors for detector calibration, ventilation flow rate, and average site meteorology.

*Allowable Values are those values assumed in accident analysis.

Addressed
in 5.02

Table 3.5-5
Radioactive Effluent Monitoring Instrumentation

Minimum
Channels
Operable Action

- 1. Gross Activity Monitors (Liquid)
 - a. Liquid Radwaste (R-18) 1 1
 - b. Steam Generator Blowdown (R-19) 1* 2
 - c. Turbine Building Floor Drains (R-21) 1 3
 - d. High Conductivity Waste (R-22) 1 1
 - e. Containment Fan Coolers (R-16) 1 3
 - f. Spent Fuel Pool Heat Exchanger A Loop (R-20A) 1+++ 3
 - g. Spent Fuel Pool Heat Exchanger B Loop (R-20B) 1+++ 3

2. Plant Ventilation

a. Without Mini-Purge

- 1. Noble Gas Activity (R-14)
 (Providing Alarm and Isolation
 of Gas Decay Tanks) 1 4
- 2. Particulate Sampler (R-13) 1 5
- 3. Iodine Sampler (R-10B or R-14A***) 1 5

b. With Mini-Purge

- 1. Noble Gas Activity (R-14) 1 4
- 2. Particulate Sampler (R-13) 1 5
- 3. Iodine Sampler (R-10B or R-14A***) 1 5
- 4. Noble Gas Activity (R-12) 1++ 8
 or
 Particulate Sampler (R-11)

Addressed in
5.0. These do
not automatically
close the purge
path

3. Shutdown Purge

- a. Noble Gas Activity (R-12) 1+ 8
- b. Particulate Sampler (R-11) 1+ 8

~~3.5-20~~

~~Amendment No. 24 25 47~~

Addressed in Chapter 5.0

	<u>Minimum Channels Operable</u>	<u>Action</u>
--	----------------------------------	---------------

This does not auto close purge ports

	C. Iodine Sampler (R-10A or R-12A***)	1+	5
4.	Air Ejector Monitor (R-15 or R-15A***)	1**	6
5.	Waste Gas System Oxygen Monitor	1	7

* Not required when Steam Generator Blowdown is being recycled (i.e. not released).

+ 15.ix

Required ^{in Modes 1-4 and for LCo 2.9.3} ~~only during shutdown purges~~ and required to sample the containment stack.

++

Required to sample containment during mini-purge operation.

**

Not required during Cold or Refueling Shutdown.

Also see Table 3.5-6.

+++

Applicable when Heat Exchanger in service.

Addressed in Chapter 5.0

3.5-20a

Amendment No. 79, 43

TABLE 3.5-5 (Continued)Table Notation

Action 1 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases from the tank may continue for up to 14 days, provided that prior to initiating a release:

1. At least two independent samples of the tank's contents are analyzed, in accordance with Specification 4.12.1.1.a, and
2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Addressed with
Chapter 5.0

Otherwise, suspend release of radioactive effluents via this pathway.

Action 2 - When Steam Generator Blowdown is being released (not recycled) and the number of channels operable is less than required by the Minimum Channels Operable requirements, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10⁻⁷ uCi/gram:

1. At least once per 8 hours when the concentration of the secondary coolant is > 0.01 uCi/gram dose equivalent I-131.
2. At least once per 24 hours when the concentration of the secondary coolant is ≤ 0.01 uCi/gram dose equivalent I-131.

Action 3 - If the number of operable channels is less than required by the Minimum Channels operable requirement, effluent releases via this pathway may continue provided that at least once per 24 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10⁻⁷ uCi/gm.

Action 4 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for isotopic activity within 24 hours or R14A is operable and readings are reviewed at least once per 8 hours.

~~3.5-21~~

TABLE 3.5-5 (Continued)

Table Notation

- Action 5 - If the number of operable channels is less than required by the Minimum Channels Operable requirements, effluent releases via this pathway may continue provided samples are continuously collected as required by Table 4.12-2 Item E with auxiliary sampling equipment.
- Action 6 - If the number of operable channels is less than required by the Minimum Channels Operable and the Secondary Activity is $\leq 1 \times 10^{-4}$ uCi/gm, effluent releases may continue via this pathway provided grab samples are analyzed for gross radioactivity (beta or gamma) at least once per 24 hours. If the secondary activity is greater than 1×10^{-4} uCi/gm, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken every 8 hours and analyzed within 24 hours.
- Action 7 - If the channel is inoperable, a sample of the gas from the in service gas decay tank shall be analyzed for oxygen content at least once every 4 hours.

Action 8 - If the number of operable channels is less than required by the Minimum Channels Operable, or at least one containment fan cooler is not operating, within 4 hour terminate the purge.

Addressed with Chapter 5.0

4

15.iii

Table 3.5-6

Radiation Accident Monitoring Instrumentation

<u>Instrument</u>	<u>Minimum Channels Operable</u>	<u>Action</u>
1. Containment Area (R-29 and R-30)	2	1
2. Noble Gas Effluent Monitors		
i. Plant Vent (R-14A)	1	1
ii. A Main Steam Line (R-31)	1	1
iii. B Main Steam Line (R-32)	1	1
iv. Containment Purge (R-12A)	1*	1
v. Air Ejector (R-15A)	1	1

Action Statements

Action 1 - With the number of operable channels less than required by the Minimum Channels Operable requirements, either restore the inoperable channel(s) to operable status within 7 days of the event, or prepare and submit a Special Report to the Commission within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

* Only when the shutdown purge flanges are removed.

15.iv

3.6.3 Containment Isolation Boundaries

3.6.3.1 With a containment isolation boundary inoperable for one or more containment penetrations, either:

Addressed w/
Chapter 3.6

- a. Restore each inoperable boundary to OPERABLE status within 4 hours, or
- b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, one closed manual valve, or a blind flange, or
- c. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

LC0333 3.6.4 Combustible Gas Control

Table 3.3-1
FU #11

3.6.4.1 16.viii When the reactor is critical, at least two independent containment hydrogen monitors shall be operable. One of the monitors may be the Post Accident Sampling System.

Cond A
Cond B

3.6.4.2 16.ix With only one hydrogen monitor operable, restore a second monitor to operable status within 30 days or be in at least hot shutdown within the next 6 hours.

Cond E
Cond G

3.6.4.3 16.viii With no hydrogen monitors operable, restore at least one monitor to operable status within 72 hours or be in at least hot shutdown within the next 6 hours.

3.6.5 Containment Mini-Purge

Whenever the containment integrity is required, emphasis will be placed on limiting all purging and venting times to as low as achievable. The mini-purge isolation valves will remain closed to the maximum extent practicable but may be open for pressure control, for ALARA, for respirable air quality considerations for personnel entry, for surveillance tests that may require the valve to be open or other safety related reasons.

Addressed with
Chapter 3.6

3.12 Movable In-Core Instrumentation

Applicability:

Applies to the operability of the movable detector instrumentation system.

Objective:

To specify functional requirements on the use of the in-core instrumentation systems for the recalibration of the excore axial off-set detection system.

Specification:

3.12.1 A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during recalibration of the excore axial off-set detection system.

SR 3.3.1.6

22i

3.12.2 Power shall be limited to 90% of rated power if the calibration requirements for excore axial off-set detection system, identified in Table 4.1-1, are not met.

SR 3.3.1.6

Basis:

(1)

The Movable In-Core Instrumentation System has four drives, four detectors, and 36 thimbles in the core. Each detector can be routed to nineteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detector channels, it is only necessary that the

Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it will compensate for an error of 10% in the excore protection system. Experience at Beznau (Switzerland) and Ginna has shown that drift due to changes in the core or instrument channels is very slight. Thus, the 10% reduction is considered to be very conservative.

Reference:

- (1) FSAR - Section 7.6

4.0 SURVEILLANCE REQUIREMENTS

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 Operational Safety Review

Applicability:

Applies to items directly related to safety limits and limiting conditions for operation:-

Objective:

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

4.1.1 Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.

LCO 3.3.1
LCO 3.3.2
LCO 3.3.4
LCO 3.3.5

4.1.2 Equipment and sampling tests shall be conducted as specified in Table 4.1-2 and 4.1-4.

4.1.3 Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check and channel calibration operations at the frequencies shown in Table 4.1-3.

LCO 3.3.3

4.1.4 Each radioactive effluent monitoring instrumentation channel shall be demonstrated operable by performing the channel check, source check, channel functional test, and channel calibration at the frequency shown in Table 4.1-5.

Addressed with Chapter 50

Basis:Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor and steam system instrumentation.

Control Room procedures require a check of the Radiation Monitoring System (RMS) panel meters and strip chart recorders for proper readout once each shift. A daily surveillance log is also maintained in the Control Room for manual entry of RMS readouts, and is independently reviewed by Health Physics supervision at least weekly.

A radiation monitor downscale failure will result in a conspicuous visual indication on the RMS panel (no audible alarm). Radiation monitor control switches are spring-returned to the "operate" mode after being turned to any other test or check mode. Therefore, together with the design features of the RMS, plant surveillance procedures ensure the continued availability of each radiation monitor to perform its intended function.

Calibration

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each refueling shutdown for the process system channels is considered acceptable.

Testing

The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hr. per channel. This is based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For the specified one month test interval, the average unprotected time is 360 hours in case of a failure occurring between test intervals. Thus, the probability of failure of one channel between test intervals is $360 \times 2.5 \times 10^{-6}$ or $.9 \times 10^{-3}$. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two-out-of-three channels is $3(.9 \times 10^{-3})^2 = 2.4 \times 10^{-6}$. This represents the fraction of time in which each three-channel system would have one operable and two inoperable channels and equals $2.4 \times 10^{-6} \times 8760$ hours per year, or (approximately) 1 minute/year.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range FU#2 (3.3.1) FU#5 (3.3.1) <i>See also Chapter 3.1</i>	S 1 (3.3.1) M 3 (3.3.1) 3 (3.3.1)	D 1 (3.3.1) Q 3 (3.3.1) 6 (3.3.1)	7 (3.3.1) P 4 (3.3.1) P 5 (3.3.1) 8 (3.3.1)	1) Heat balance calculation** 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset** 4) High setpoint (<109% of rated power) 5) Low setpoint (<25% of rated power)
2. Nuclear Intermediate Range FU#3 (3.3.1) <i>Mode 6 Requirements Addressed w/Chapter 3.9</i>	S 1 (3.3.1)	N.A. R 10 (3.3.1)	P 8 (3.3.1)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3. Nuclear Source Range FU#4 (3.3.1)	S 1 (3.3.1)	N.A. R 10 (3.3.1)	P 8 (3.3.1)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4. Reactor Coolant Temperature FU#3 (3.3.3) FU#5 (3.3.1) FU#6 (3.3.1) FU#11 (3.3.2) <i>See also Chapter 3.1</i>	S 1 (3.3.1)	R 10 (3.3.1)	Q 7 (3.3.1) M 1 (2)	1) Overtemperature-Delta T 2) Overpower - Delta T
5. Reactor Coolant Flow FU#9 (3.3.1)	S 1 (3.3.1)	R 10 (3.3.1)	M 7 (3.3.1) Q 7 (3.3.1)	
6. Pressurizer Water Level UH#8 (3.3.1)	S 1 (3.3.1)	R 10 (3.3.1)	M 7 (3.3.1) Q 7 (3.3.1)	
7. Pressurizer Pressure FU#7 (3.3.1)	S 1 (3.3.1)	R 10 (3.3.1) 5 (3.3.2) 6 (3.3.2)	M 7 (3.3.1) Q 2 (3.3.2)	
8. 4 KV Voltage & Frequency FU#1, d (3.3.2) FU#11; IZ (3.3.1) FU#6.c (3.3.2)	N.A.	R 10 (3.3.1) 5 (3.3.2)	M 7 (3.3.1) Q 3 (3.3.2)	<u>Reactor Protection circuits only</u>
9. Rod Position Indication	S (1, 2)	N.A.	M	1) With step counters 2) Log rod position indications each 4 hours when rod deviation monitor is out of service

ADD Function #10, "RCP Breaker Position"

ADD Function #14, "SI Input from SFAS"

By means of the movable in-core detector system.

** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

SR 3.3.1.2 NOTE

TABLE 4.1-1 (Continued)

28.i.a
28.i.b

Addressed in Chapter 3.1

Channel Description

Check

Calibrate Test

Remarks

10. Rod Position Bank Counters

S(1,2)

N.A.

N.A.

1) With rod position indication
2) Log rod position indications each 4 hours when rod deviation monitor is out of service

FU # 1,2,3,3.3
FU # 13(3.3.1)
FU # 5.b and 6.c (3.3.2)
12. Charging Flow

S^{1(3.3.1)}
N.A.

R^{10(3.3.1)}
R^{5(3.3.2)}

M^{7(3.3.1)}
N.A.

LC0333
FU # 14
13. Residual Heat Removal Pump Flow

N.A.

R^{2(3.3)}

N.A.

14. Boric Acid Storage Tank Level

D

R

N.A.

Note 4

15. Refueling Water Storage Tank Level

N.A.

R

N.A.

Addressed with Chapter 3.4 & 3.5

16. Volume Control Tank Level

N.A.

R

N.A.

17. Reactor Containment Pressure

D^{1(3.3.2)}

R^{5(3.3.2)}

M^{9(1)(3.3.2)}

1) Isolation Valve signal

28.i.c

FU # 1.c (3.3.2)
FU # 2.c (3.3.2)
18. Radiation Monitoring System

D

R

M

Area Monitors R1 to R9, System Monitor R17

19. Boric Acid Control

N.A.

R

N.A.

20. Containment Drain Sump Level

N.A.

R

N.A.

Addressed with Chapter 3.4 & 3.5

FU # 4
21. Valve Temperature Interlocks

N.A.

N.A.

R

FU # 1,2,3,4,3.c
22. Pump-Valve Interlock

R

N.A.

N.A.

FU # 14 (3.3.1)
23. Turbine Trip Set-Point

N.A.

R^{10(3.3.1)}

M^{13(3.3.1)}

1) Block Trip

28.i.c

24. Accumulator Level and Pressure

S

R

N.A.

Addressed with Chapter 3.4 & 3.5

Amendment No. 7-57

4.1-6

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
25. Containment Pressure	(S) ¹ (3.3.2)	(R) ⁵ (3.3.1)	(M) ² (3.3.2)	Narrow range containment pressure (-3.0/ +3 psig) excluded
26. Steam Generator Pressure FU# 1.0 (3.3.2)	(S) ¹ (3.3.1)	(R) ⁵ (3.3.2)	(M) (3.3.2)	
27. Turbine First Stage Pressure	S	R	(M) (3.3.2)	
28. Emergency Plan Radiation Instruments	M	R	M	
29. Environmental Monitors	M	NA	NA	
30. Loss of Voltage/Degraded Voltage 480 Volt Safeguards Bus LCO 3.3.4	NA	(R) ² (3.3.4)	(M) ¹ (3.3.4)	
31. Trip of Main Feedwater Pumps FU# 6.4 (3.3.2)	NA	NA	(R) ⁴ (3.3.2)	
32. Steam Flow FU# 4.3 and 4.e (3.3.2)	(S) ¹ (3.3.2)	(R) ⁵ (3.3.2)	(M) (3.3.2)	
33. T _{AVI} FU# 4.d (3.3.2)	(S) ¹ (3.3.2)	(R) ⁵ (3.3.2)	(M) (3.3.2)	
34. Chlorine Detector, Control Room Air Intake	NA	R	M	
35. Ammonia Detector, Control Room Air Intake	NA	R	M	
LCO 3.3.5 36. Radiation Detectors, Control Room Air Intake	NA	(R) ² (3.3.5)	(M) (3.3.5)	
LCO 3.3.3 37. Reactor Vessel Level Indication System	(M) ¹ (3.3.3)	(R) ² (3.3.3)	NA	
38a. Trip Breaker Logic Channel Testing FU# 17 (3.3.1)	NA	NA	(M) ⁵ (3.3.1)	
38b. Trip Breaker Logic Channel Testing FU# 1 (3.3.1)	NA	NA	(R) ¹¹ (3.3.1)	

28.i.a
28.i.b

28.i.c

28.i.g

Note 1, 2 and 3

Note 1 28.i.c

TABLE 4.1-1 (Continued)

Channel Description	Check	Calibrate Test	Remarks
FU #15 FU #16 (3.3.1) 39. Reactor Trip Breakers	N.A.	N.A. (M) (3.3.1)	Function test - Includes independent testing of both undervoltage and shunt trip attachment of reactor trip breakers. Each of the two reactor trip breakers will be tested on alternate months. (28.i.a) (28.i.b) (28.i.c)
Fu #1 (3.3.1) 40. Manual Trip Reactor	N.A.	N.A. (R) (3.3.1)	Includes independent testing of both undervoltage and shunt trip circuits. The test shall also verify the operability of the bypass breaker.
FU #15 (3.3.1) 41a. Reactor Trip Bypass Breaker	N.A.	N.A. (M) (3.3.1)	Using test switches in the reactor protection rack manually trip the reactor trip bypass breaker using the shunt trip coil.
FU #1 (3.3.1) 41.b Reactor Trip Bypass Breaker	N.A.	N.A. (R) (3.3.1)	Automatically trip the undervoltage trip attachment.

SR 3.3.1.14 through SR 3.3.1.18 (28.i.f)

NOTE 1: Logic trains will be tested on alternate months corresponding to the reactor trip breaker testing. Monthly logic testing will verify the operability of all sets of reactor trip logic actuating contacts on that train (See Note 3). Refueling shutdown testing will verify the operability of all sets of reactor trip actuating contacts on both trains. In testing, operation of one set of contacts will result in a reactor trip breaker trip; the operation of all other sets of contacts will be verified by the use of indication circuitry. (28.i.c)

NOTE 2: Testing shall be performed monthly, unless the reactor trip breakers are open or shall be performed prior to startup if testing has not been performed within the last 30 days.

NOTE 3: The source range trip logic may be excluded from monthly testing provided it is tested within 30 days prior to startup.

NOTE 4: When BAST is required to be operable.

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	<u>Test</u>	<u>Frequency</u>
1.	Reactor Coolant Chemistry Samples Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2.	Reactor Coolant Boron	Boron Concentration Weekly
3.	Refueling Water Storage Tank Water Sample	Boron Concentration Weekly
Addressed with Chapter 3.4 & 3.5		
4.	Boric Acid Storage Tank	Boron Concentration Twice/Week ⁽⁶⁾
5.	Control Rods	Rod drop times of all full length rods After vessel head removal and at least once per 18 months (1)
6a.	Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system Monthly
Addressed with Chapter 3.1		
6b.	Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur Each Refueling Shutdown
7.	Pressurizer Safety Valves	Set point Each Refueling Shutdown
Addressed with Chapter 3.4		
8.	Main Steam Safety Valves	Set point Each Refueling Shutdown
Addressed with Chapter 3.5		
9.	Containment Isolation Trip	Functioning Each Refueling Shutdown
10.	Refueling System Interlocks	Functioning Prior to Refueling Operations
Addressed in Chapter 3.9		

SR 23.2.7

Addressed with Chapter 3.2

	<u>Test</u>	<u>Frequency</u>	
11.	Service Water System	Functioning	Each Refueling Shutdown
12.	Fire Protection Pump and Power Supply	Functioning	Monthly
13.	Spray Additive Tank	NaOH Concent	Monthly
14.	Accumulator	Boron Concentration	Bi-Monthly
15.	Primary System Leakage	Evaluate	Daily
16.	Diesel Fuel Supply	Fuel Inventory	Daily
17.	Spent Fuel Pit	Boron Concentration	Monthly
18.	Secondary Coolant Samples	Gross Activity	72 hours (2) (3)
19.	Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown

Addressed with Chapter 3.6

Addressed with Chapter 3.4 & 3.5

Addressed with Chapter 3.2

Addressed with Chapter 3.7 & 4.0

Addressed with Chapter 3.4

20.u.h

Notes:

Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

- (2) Not required during a cold or refueling shutdown.
- (3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.
- (4) When BAST is required to be operable.

TABLE 4.1-3

Accident Monitoring Instrumentation Surveillance Requirements

<u>Instrument</u>	<u>Channel Check</u>	<u>Channel Calibration</u>	<u>Channel Test</u>
1. Pressurizer Water Level (1)	see Table 4.1-1	see Table 4.1-1	NA
2. Auxiliary Feedwater Flow Rate (4)	see Section 4.8.1	R	NA
3. Reactor Coolant System Subcooling Margin Monitor (2)	M	R	NA
4. Pressurizer PORV Position Indicator (primary detector) (3)	M	NA	R
5. Pressurizer PORV Position Indicator (RTD - backup detector) (3)	M	R	NA
6. PORV Block Valve Position Indicator (1)	M	NA	R
7. Pressurizer Safety Valve Position Indicator (primary detector) (3)	M	R	NA
8. Pressurizer Safety Valve Position Indicator (RTD - backup detector) (3)	M	R	NA
9. Containment Pressure	M	R	NA
10. Steam Generator Water Level - Wide Range	M	R	NA
11. Containment Water Level (Narrow Range, Sump A)	M	R	NA
12. Containment Water Level (Wide Range, Sump B)	M	R	NA
13. Core Exit Thermocouples	M	R	NA
14. Containment Area High Range Radiation (R-29 and R-30) (5)	M	R	M

- (1) Emergency Power Supply Requirements for Pressurizer Level Indicators - NUREG 0578 Item 2.1.1
(2) Instrumentation for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.1
(3) Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 item 2.1.3.a
(4) Auxiliary Feedwater Flow Indication to Steam Generator NUREG 0578 item 2.1.7.b
(5) Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG 0737

28.iii.a

Replace w/ LCO 3.3.3,
Table 3.3.3-1,
Functions #1 - #26

Amendment No. 2

Add. with
Chapter 3.4

TABLE 4.1-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE
AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>
1. Gross Activity Determination (beta-gamma) (1)	At least once per 72 hours	Above cold shutdown
2. Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days	Above 5% reactor power
3. Radiochemical for \bar{E} Determination (2)	1 per 6 months (3)	Above 5% reactor power
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 8 hours, whenever the I-131 equivalent activity exceeds the limit of 3.1.4.1.b	As required by Specification 3.1.4.3.c*
	b) One sample between 2 and 10 hours following a reactor power change exceeding 15 percent within a 1-hour period	Hot shutdown or above

- (1) A gross radiactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units $\mu\text{Ci/gm}$. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled.
- (2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of \bar{E} .
- (3) Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.

* Except at refueling shutdown, sampling shall be continued until the activity of the reactor coolant system is restored to within its limits.

Addressed with Chapter 5.0

Table 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

Instrument	Channel Check	Source Check	Functional Test	Channel Calibration
1. Gross Activity Monitor (Liquid)				
a. Liquid Rad Waste (R-18)	D(7)	M(4)	Q(1)	R(5)
b. Steam Generator Blowdown (R-19)	D(7)	M(4)	Q(1)	R(5)
c. Turbine Building Floor Drains (R-21)	D(7)	M(4)	Q(1)	R(5)
d. High Conductivity Waste (R-22)	D(7)	M(4)	Q(1)	R(5)
e. Containment Fan Coolers (R-16)	D(7)	M(4)	Q(2)	R(5)
f. Spent Fuel Pool Heat Exchanger A Loop (R-20A)	D(7)	M(4)	Q(2)	R(5)
g. Spent Fuel Pool Heat Exchanger B Loop (R-20B)	D(7)	M(4)	Q(2)	R(5)
Plant Ventilation				
a. Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D(7)	M	Q(1)	R(5)
b. Particulate Sampler (R-13)	W(7)	N.A.	N.A.	R(5)
c. Iodine Sampler (R-10B and R-14A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3. Containment Purge				
a. Noble Gas Activity (R-12)	SR 3.3.6.1 D(7)	PR	SR 3.3.2.3 Q(1)/R	SR 3.3.6.4 R(5)
b. Particulate Sampler (R-11)	28.v.d D W(7)	N.A.	Q(1)/R	R(5)
c. Iodine Sampler (R-10A and R-12A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
4. Air Ejector Monitor (R-15 and R-15A)	D(7)	M	M(2)	R(5)
5. Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)
6. Main Steam Lines (R-31 and R-32)	M	N.A.	Q	R

See also Chapter 3.4

TABLE 4.1-5 (Continued)

TABLE NOTATION

28.v.c

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm occur if any of the following conditions exist:

- 1. Instrument indicates measured levels above the alarm and/or trip setpoint.
- 2. Power failure.

(2) The Channel Functional Test shall also demonstrate that control room alarm occurs if any of the following conditions exist:

- 1. Instrument indicates measured levels above the alarm setpoint.
- 2. Power failure.

(3) The Channel Calibration shall include the use of standard gas samples containing a nominal:

- 1. Zero volume percent oxygen; and
- 2. Three volume percent oxygen.

(4) This check may require the use of an external source due to high background in the sample chamber.

(5) Source used for the Channel Calibration shall be traceable to the National Bureau of Standards (NBS) or shall be obtained from suppliers (e.g. Amersham) that provide sources traceable to other officially-designated standards agencies.

28.v.c

(6) Flow rate for main plant ventilation exhaust and containment purge exhaust are calculated by the flow capacity of ventilation exhaust fans in service and shall be determined at the frequency specified.

(7) Applies only during releases via this pathway. 28.ii.d

See Chapter 5.0

See Chapter
3.6

the tendon containing 6 broken wires) shall be inspected. The accepted criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

- a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.
- b. Before reseating, a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

- 4.4.5.1 Each containment isolation valve shall be demonstrated to be OPERABLE in accordance with the Ginna Station Pump and Valve Test program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

- 4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

Table 3.3.2-1
F.V. 3a, b, c

- 4.4.6.2 The response time of each containment isolation valve shall be demonstrated to be within its limit at least once per 18 months. The response time includes only the valve travel time for those valves which the safety analysis assumptions take credit for a change in valve position in response to a containment isolation signal.

See Chapter
3.6

4.4.7 Containment Hydrogen Monitors

4.4.7.1 Demonstrate that two hydrogen monitors are operable at least daily by verifying that the unit is on or in standby. monthly

31.x

4.4.7.2 At least once per quarter perform a channel calibration using two sample gases containing known concentrations of hydrogen. 24 months

31.x

Basis:

The containment is designed for an accident pressure of 60 psig. ⁽¹⁾ While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure. The maximum temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286° F.

See Chapter
3.7

4.8.8 At least once per 18 months during shutdown:

- a. Verify that each automatic valve in the flow path for each auxiliary feedwater pump actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.
- b. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

4.8.9 Each instrumentation channel shall be demonstrated operable by the performance of the Channel Check, Channel Calibration, and Channel Functional Test operations for the modes and at the frequencies shown in Table 4.1-1.

Table 3.3.2-1
F.V.#6

4.8.10 ~~The response time of each pump and valve required for the operation of each "train" of auxiliary feedwater shall be demonstrated to be within the limit of 10 minutes at least once per 18 months.~~

35.vii

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet minimum required flowrates. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.⁽¹⁾ Proper functioning of the steam turbine admission valve and the feedwater pumps start will demonstrate the integrity of the steam driven pump.⁽²⁾

Monthly testing of the standby auxiliary feedwater pumps by supplying water from a condensate supply tank to the steam generators will verify their ability to meet minimum required flowrates.

The standby auxiliary feedwater pumps would be used only if all three auxiliary feedwater pumps were unavailable.⁽³⁾ One of the two standby pumps would be sufficient to meet decay heat removal requirements. Proper functioning of the suction valves from the service water system, the discharge valves, and the crossover valves will demonstrate their operability. The operability of the standby auxiliary feedwater pump flow paths between the pumps and the steam generators is demonstrated using water from the test tank. Testing of the auxiliary feedwater pumps using their primary source of water supply will verify the operability of the auxiliary feedwater flow path.

Verification of correct operation will be made both from instrumentation within the main control room and by direct visual observation of the pumps.

3.1 Reactor Coolant System

Applicability:

Applies to the operating status of the Reactor Coolant System when fuel is in the reactor.

Objective:

To specify those conditions of the Reactor Coolant System which must be met to assure safe reactor operation.

Specification:

3.1.1 Operational Components

3.1.1.1 Reactor Coolant Loops

- a. When the reactor power is above 130 MWT (8.5%), both reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation.
- b. If the conditions of 3.1.1.1.a are not met, then immediate power reduction shall be initiated under administrative control. If the shutdown margin meets the one loop requirements of Figure 3.10-2, then the power shall be reduced to less than 130 MWT. If the one loop shutdown margin of Figure 3.10-2 is not met, the plant shall be taken to the hot shutdown condition and the one loop shutdown margin shall be met.
- c. Except for special tests, when the RCS temperature is at or above 350°F with the reactor power less than or equal to 130 MWT (8.5%), at least one reactor coolant loop and its associated steam generator and reactor coolant pump shall be in

LCO 3.4.4

LCO 3.4.4
LCO 3.4.5

(6.ii)
(6.i)

LCO 3.4.5
(6.ii)

operation. The other loop and its associated steam generator must be operable so that heat could be removed via natural circulation. However, both reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

- d. If the conditions of 3.1.1.1.c are not met, then
- (i) if one loop is in operation, but the other loop is not operable, restore the inoperable loop to operable status within 72 hours or take the plant to the hot shutdown condition and reduce the RCS temperature to less than 350°F within the next 12 hours, or
 - (ii) if neither loop is in operation suspend all operations involving a reduction in boron concentration in the Reactor Coolant System and immediately initiate corrective action to return a coolant loop to operation.

LCO 3.4.5

(6.ii)

LCO 3.4.3

(6.ii)

- e. When the RCS temperature is less than 350°F, at least two of the following coolant loops shall be operable:

LCO 3.4.6

LCO 3.4.7

LCO 3.4.2

(6.vii)

- (i) reactor coolant loop A and its associated steam generator and reactor coolant pump.
- (ii) reactor coolant loop B and its associated steam generator and reactor coolant pump.

- (iii) residual heat removal loop A.*
- (iv) residual heat removal loop B.*

LCO 3.4.8
 LCO 3.4.7
 LCO 3.4.6

(6.iii)
 (6.vi)

f. ~~Except during steam generator crevice cleaning operations~~ at least one of the coolant loops listed in paragraph 3.1.1.1.e shall be in operation while RCS temperature is less than 350°F. However, both reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

LCO 3.4.7
 LCO 3.4.6
 LCO 3.4.8

(6.iv)

g. If the conditions of 3.1.1.1.e are not met, immediately initiate corrective action to return the required loops to operable status, and if not in cold shutdown already, be in cold shutdown within 24 hours.

LCO 3.4.6
 LCO 3.4.7
 LCO 3.4.8

h. If the conditions of 3.1.1.1.f are not met, then suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*The preferred or emergency power source may be inoperable while in cold shutdown.

6.vii

i. At least one reactor coolant pump or the residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

6.viii

j. At least one reactor coolant pump shall be in operation for a planned transition from one Reactor Operating Mode to another involving an increase in the boron concentration of the reactor coolant, except for emergency boration.

LCO 3.4.7
LCO 3.4.6

6.v

Reference to P11C

k. A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures $\leq 330^{\circ}\text{F}$ unless 1) the pressurizer water volume is less than 324 cubic feet (38% level) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3.1.1.2 Steam Generator

6.xv

a. The temperature difference across the tube sheet shall not exceed 100°F .

3.1.1.3 Safety Valves

6.xii

a. During cold shutdown or refueling when the reactor head is bolted on the vessel, at least one pressurizer code safety valve shall be operable with a lift setting of $2485 \text{ psig} \pm 1\%$.

6.xiii

b. If the conditions of 3.1.1.3.a are not met, immediately suspend all operations involving positive reactivity changes and place an operable RHR loop into operation in the shutdown cooling mode.

c. Whenever the reactor is at or above an RCS temperature of 350°F, both pressurizer code safety valves shall be operable with a lift setting of 2485 psig $\pm 1\%$.

LCO 3.4.10

G.Xvii

d. If one pressurizer code safety valve is not operable while the reactor is at or above an RCS temperature of 350°F, then either restore the inoperable valve to operable status within 15 minutes or be in at least hot shutdown within 6 hours and below an RCS temperature of 350°F within an additional 6 hours.

LCO 3.4.10.

G.Xvi

3.1.1.4 Relief Valves

a. Both pressurizer power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor is at or above an RCS temperature of 350°F, or

(i). with one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and below an RCS temperature of 350°F within the following 6 hours, or

(ii) with one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s);

LCO 3.4.11

G.Xiii

LCO 3.4.11

G.Xiv

otherwise, be in at least hot shutdown within the next 6 hours and below an RCS temperature of 350°F within the following 6 hours.

3.1.1.5 Pressurizer

LCO 3.4.3

G.ix

- a. Whenever the reactor is at or above an RCS temperature of 350°F the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hrs. or have the reactor below an RCS temperature of 350°F and the RHR system in operation within an additional 6 hrs.

G.x

- ~~b. This requirement shall not apply during performance of RCS hydro test provided the test is completed and the pressurizer is operable per 3.1.1.5a within 16 hours.~~

3.1.1.6 Reactor Coolant System Vents

LCO 3.4.1

G.xiii

G.xiv

G.xi

- a. When the reactor is at hot shutdown or critical, at least one reactor coolant system vent path consisting of two valves in series shall be operable and closed* at each of the following locations:

~~1. Reactor Vessel head~~

2. Pressurizer steam space

*The PORV block valve is not required to be closed but must be operable if the PORV is capable of being opened.

b. With one or more vents at the above reactor coolant system vent path locations inoperable, startup may commence and/or power operation may continue provided at least one vent path is operable and the inoperable vent paths are maintained closed with motive power removed from the valve actuator of all the valves in the inoperable vent paths. If the requirements of 3.1.1.6a are not met within 30 days, be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

260 3.4.11

G.xi

G.xiii

G.xiv

c. With all of the above reactor coolant system vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 72 hours or be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

260 3.4.11

G.xi

G.xiii

G.xiv

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

operations and anticipated transients. Heat transfer analyses⁽¹⁾ show that reactor heat equivalent to 130 MWT (8.5%) can be removed by natural circulation alone. Therefore operation with one operating reactor coolant loop while below 130 MWT provides adequate margin.

The specification permits an orderly reduction in power if a reactor coolant pump is lost during operation between 130 MWT and 50% of rated power.⁽²⁾ Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than one which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase.

When the reactor coolant system average temperature is above 350°F, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require one loop be in operation and the other loop be capable of removing heat via natural circulation.

When the reactor coolant system average temperature is between 200°F and 350°F or while in cold shutdown, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity

changes in the reactor. Mixing of the reactor coolant will be sufficient to prevent a sudden increase in reactivity if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant. When the boron concentration of the reactor coolant system is to be increased, the process must be uniform to prevent sudden reactivity increases in the reactor during subsequent startup of the reactor coolant pumps. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump is running while the change is taking place. Emergency boration without a reactor coolant pump in operation is not prohibited by this specification.

Prohibiting reactor coolant pump starts without a large void in the pressurizer or without a limited RCS temperature differential will prevent RCS overpressurization due to expansion of cooler RCS water as it enters a warmer steam generator. A 38% level in the pressurizer will accommodate the swell resulting from a reactor coolant pump start with a RCS temperature of 140°F and steam generator secondary side temperature of 340°F, or the maximum temperature which usually exists prior to cooling the reactor with the RHR system.

Temperature requirements for the steam generator correspond with measured NDT for the shell and allowable thermal stresses in the tube sheet.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The requirement that 100 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown and during cooldown. (3)

Reactor Coolant System Vents

Reactor Coolant System Vents are provided to exhaust noncondensable gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head and one from the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

References

- (1) FSAR Section 14.1.6
- (2) FSAR Section 7.2.3
- (3) Letter from L.D. White, Jr. to D. L. Ziemann, USNRC, dated October 17, 1979

3 1.2

Heatup and Cooldown Limit Curves for Normal Operation

3.1.2.1

The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2 for the first 21.0 effective full power years.

LCO 3.4.3

(7.i)

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. The heatup and cooldown rates shall not exceed 60°F/hr and 100°/hr, respectively. Limit lines for cooldown rates between those presented may be obtained by interpolation.

LCO 3.4.3

(7.i)

b. Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. The limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using methods discussed in the Basis Section.

LCO 3.4.3

(7.ii)

c. If the limits on Figures 3.1-1 and 3.1-2 are exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; and either

- 1) within ⁷²8 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System and determine that the Reactor Coolant System remains acceptable for continued operation, or

LCO 3.4.3

(7.iii)

LCO 3.4.3

2) within 6 hours be in at least HOT SHUTDOWN, and within the next 30 hours reduce RCS temperature and pressure to less than 200°F and 500 psig, respectively.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator vessel is below 70°F.

7.iv

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

7.v

Basis:

Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code, Reference (1), and ASTM E185, Reference (2), and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code, Reference (3) and the calculation methods described in Reference (4). The results are reported in Reference (5) for Capsule T.

The heatup and cooldown curves are based on nominal pressure-temperature indications. Sufficient conservatism exists in the algorithm from which the curves were derived to account for instrument uncertainties.

TEXT DELETED

The temperature requirements for the steam generator corresponds with the measured NDT for the shell of the steam generator.

A temperature difference of 320°F between the pressurizer and reactor coolant system maintains stresses within the pressurizer spray nozzle below design limits.

- (1) ASME Boiler and Pressure Vessel Code Section III (Summer 1965)
- (2) ASTM E185 Surveillance Tests on Structural Materials in Nuclear Reactors
- (3) ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda (note Code Class 1514)
- (4) Regulatory Guide 1.99, Rev. 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials."
- (5) Westinghouse Report, "Rochester Gas and Electric Reactor Vessel Life Attainment Plan", dated March 1990.

7.1

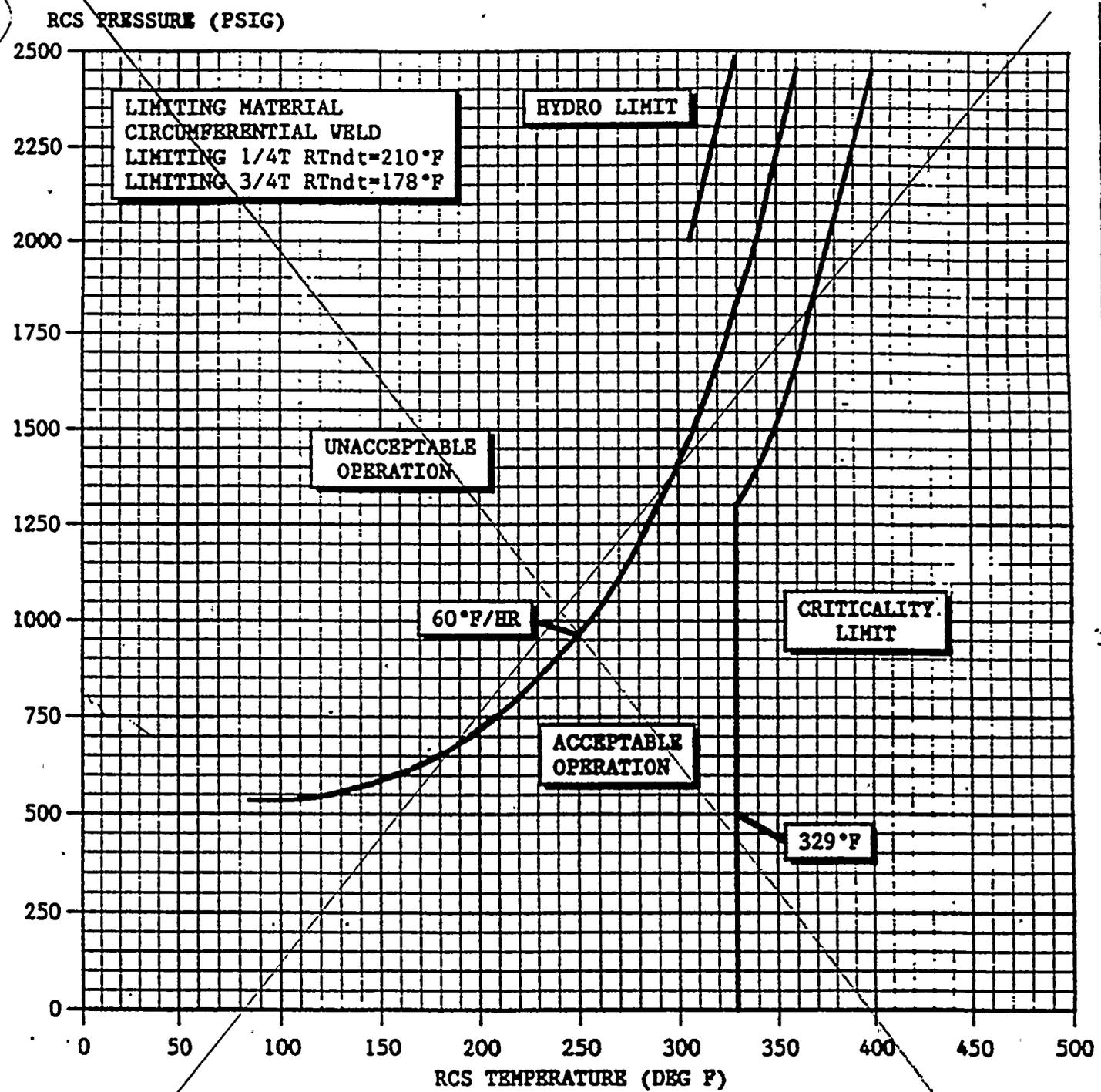


Figure 3.1-1: Ginna Reactor Vessel Heatup Limitations Applicable for the first 21 EFPY using Reg Guide 1.99, Rev. 2

7.1

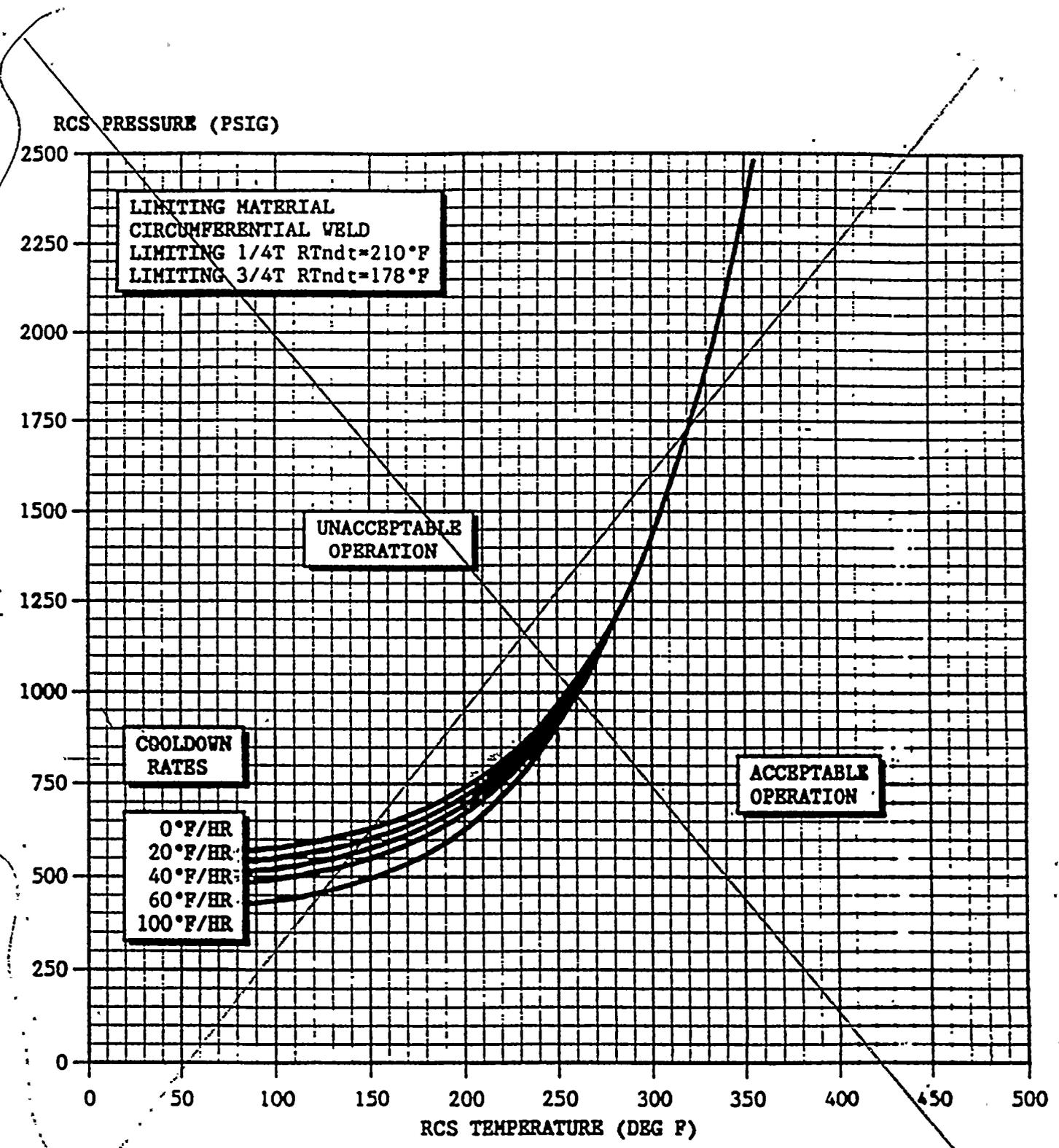


Figure 3.1-2: Ginna Reactor Vessel Cooldown Limitations Applicable for the first 21 EPFY using Reg Guide 1.99, Rev. 2.

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at a temperature below 500°F, and if the moderate temperature coefficient is more positive than

- a. 5 pcm/°F (below 70 percent of rated thermal power)
- b. 0 pcm/°F (at or above 70 percent of rated thermal power)

8.i 8.iv

LCO 3.4.2

3.1.3.2 ~~In no case shall the reactor be made critical above and to the left of the criticality limit line shown on Figure 3.1-1 of these specifications.~~

8.ii

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

LCO 3.4.2

8.iii

Basis

~~Previous safety analyses have assumed that for Design Basis Events (DBE) initiated from the hot zero power or higher power condition, the moderator temperature coefficient (MTC) was either zero or negative. (1)(2) Beginning in Cycle 14, the safety analyses have assumed that a maximum MTC of +5 pcm/°F can exist up to 70% power. Analyses have shown that the design criteria can be satisfied for the DBE's with this assumption. (3) At greater than 70% power the MTC must be zero or negative.~~

The limitations on MTC are waived for low power physics tests to permit measurement of the MTC and other physics design parameters of interest. During these tests special operating precautions will be taken.

The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase in moderator temperature or a decrease of coolant pressure.

Reference

- (1) FSAR Table 3.2.1-1
- (2) FSAR Figure 3.2.1-8
- (3) Safety Evaluation for R.E. Ginna Transition to 14 x 14 Optimized Fuel Assemblies, Westinghouse Electric Corporation, November 1983.

3.1.4 Maximum Coolant Activity

Specifications

3.1.4.1 Whenever the reactor is critical or the reactor coolant average temperature is greater than 500°F:

LCO 3.4.16
Condition C

a. The total specific activity of the reactor coolant shall not exceed $84/\bar{E}$ $\mu\text{Ci/gm}$, where \bar{E} is the average beta and gamma energies per disintegration in Mev.

LCO 3.4.16
Condition A

b. The I-131 equivalent of the iodine activity in the reactor coolant shall not exceed 0.2 $\mu\text{Ci/gm}$.

c. The I-131 equivalent of the iodine activity on the secondary side of a steam generator shall not exceed 0.1 $\mu\text{Ci/gm}$. *See Chapter 3.7*

LCO 3.4.16
Condition C

3.1.4.2 If the limit of 3.1.4.1.a is exceeded, then be subcritical with reactor coolant average temperature less than 500°F within .8 hours.

LCO 3.4.16
Condition A

3.1.4.3 a. If the I-131 equivalent activity in the reactor coolant exceeds the limit of 3.1.4.1.b but is less than the allowable limit shown on Figure 3.1.4-1, operation may continue for up to 168 hours.

The reactor may be taken critical or reactor coolant average temperature may be increased above a 500°F with the I-131 equivalent activity greater than the limit of 3.1.4.1.b as long as the provisions of this paragraph are met.

- b. If the I-131 equivalent activity exceeds the limit of 3.1.4.1.b for more than 168 hours during one continuous time interval or exceeds the limit shown on Figure 3.1.4-1, be subcritical with reactor coolant average temperature less than 500°F within 8 hours.
- c. If the I-131 equivalent activity exceeds the limit of 3.1.4.1.b, then perform sampling and analysis as required by Table 4.1-4, item 4a, until the activity is reduced to less than the limit of 3.1.4.1.b.

LCO 3.4.16

LCO 3.4.16

- 3.1.4.4 If the limit of 3.1.4.1.c is exceeded, then be at hot shutdown within 8 hours and in cold shutdown within the following 32 hours.

LCO 3.4.16

(9.1)

Basis:

The total activity limit for the primary system corresponds to operation with the plant design basis of 1% fuel defects.⁽¹⁾
Radiation shielding and the radioactive waste disposal systems

(3.1-22)

were designed for operation with 1% defects⁽²⁾. The limit for secondary iodine activity is conservatively established with respect to the limits on primary system iodine activity and primary-to-secondary leakage (Specification 3.1.5.2). If the activity should exceed the specified limits following a power transient the major concern would be whether additional fuel defects had developed bringing the total to above 1% defects. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the volume control tank gases to the gas decay tanks.

The specified activity limits provide protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the analysis of a steam generator tube rupture accident.⁽³⁾

The 500°F temperature in the specification corresponds at saturation to 681 psia, which is below the set point of the secondary side relief valves. Therefore, potential primary to secondary leakage at a temperature below 500°F can be contained by closing the steam line isolation valves.

References:

- (1) FSAR Table 9.2-5
- (2) FSAR Section 11.1.3
- (3) Letter dated September 24, 1981 from Dennis M. Crutchfield, USNRC, to John E. Maier, RG&E.

Moved to
Figure 3.1.4-1

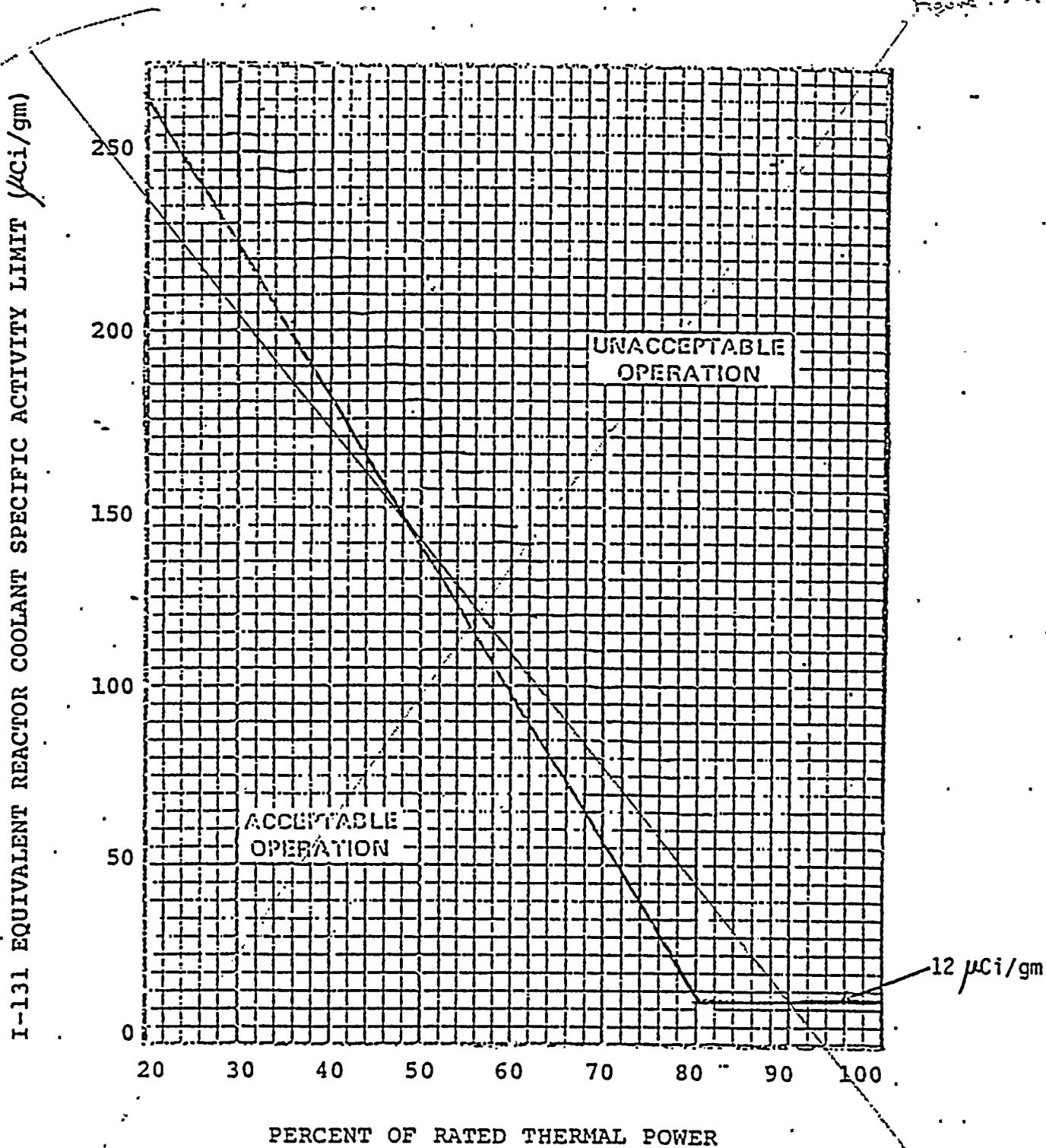


Figure 3.1.4-1

I-131 Equivalent Reactor Coolant Specific Activity Limit Versus Percent of Rated Thermal Power

3.1.5 RCS Leakage3.1.5.1 Detection Systems

3.1.5.1.1 With an RCS temperature greater than 350°F, two of the following leak detection systems, including one system sensitive to radioactivity, shall be in operation.

LCO 3.4.15

(10.i)

(10.ii)

(10.iii)

- a. The containment air particulate monitor
- b. The containment radiogas monitor
- c. The containment atmosphere humidity detector
- d. The containment water inventory monitoring system

3.1.5.1.2 When a system sensitive to radioactivity is not operable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once every 24 hours. Otherwise be in hot shutdown within the next 6 hours and have the RCS temperature less than 350°F within the following 6 hours.

LCO 3.4.15

(10.ii)

3.1.5.2 RCS Leakage Limits

3.1.5.2.1 With the RCS temperature at or above 350°F, RCS leakage shall be limited to:

LCO 3.4.13

- a. No leakage, if known to be through an RCS pressure boundary such as a pipe, vessel or valve body,
- b. 10 gpm from a known leakage source other than the above,
- c. 1 gpm from an unidentified leakage source,
- d. .1 gpm tube leakage in one steam generator when averaged over 24 hours.

3.1.5.2.2 If the limits specified above are exceeded the following action is required.

- CCO 3.4.13
- a. With any RCS pressure boundary leakage, as defined in 3.1.5.2.1.a, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F in the following 6 hours.
- CCO 3.4.13
- b. With leakage in excess of 3.1.5.2.1 b or c, reduce leakage rate to within limits within 4 hours or be in hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.
- CCO 3.4.13
- c. With steam generator tube leakage in excess of 3.1.5.2.1d, be at hot shutdown within 6 hours and at an RCS temperature less than 350°F within the following 6 hours. If more than six months have elapsed since the last steam generator inspection, perform an inspection in accordance with the requirements of Technical Specification 4.2.

Basis

Water inventory balances, monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether or not it is from the reactor coolant system pressure boundary, can be a serious problem with respect to in-plant radioactivity contamination or it could develop into a still more serious problem if the leakage rate is of sufficient magnitude to effect cooling of the reactor core; and, therefore, first indications of such leakage should be investigated as soon as practicable.

If leakage is to the containment, its presence may be indicated by one or more of the following methods:

- a. The containment air particulate monitor is sensitive to low leak rates. The rate of leakage to which the instrument is sensitive is 0.013 gpm within twenty minutes, assuming the presence of corrosion product activity.
- b. The containment radiogas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to greater than 10 gpm.
- c. The humidity detector provides a backup to a. and b. The sensitivity range of this instrument is from approximately 2 gpm to 10 gpm.
- d. A leakage detection system which determines leakage from water and steam systems within the containment collects and measures moisture condensed from the containment atmosphere by cooling coils of the main recirculation units. This system provides a dependable and accurate means of determining total leakage, including leaks from the cooling coils. This system can detect leakage from approximately 1/2 gpm to 10 gpm.

Indication of leakage from the above sources should be cause for an investigation and could require a containment entry and limited inspection at power of the reactor coolant system. Visual inspection procedures, i.e., looking for steam, floor wetness or boric acid crystalline formations, would be used.

It should be noted that detection systems sensitive to radioactivity will have an indication that is sensitive to the coolant activity and the location of the leak as well as the leak rate. Also since leakage directly into the containment could be from a variety of sources, such as the component cooling system, the service water system, the secondary system, the reactor make-up water system, the chemical and volume control system, the seal injection system, the sampling system, as well as the primary coolant system, an increase in containment air moisture or sump actuation does not necessarily mean a primary system leak. Water inventory balances, liquid waste activities and tritium content can all be used in determining the nature of a leak inside the containment.

If leakage is to another system, it will be detected by the plant radiation monitors and/or water inventory control.

When the source of leakage has been investigated, the situation can be evaluated to determine if operation can be continued safely. This evaluation will be within the criteria of this specification.

- a. A leak of any magnitude in a pipe, vessel, or valve body in the coolant system pressure boundary compromises the integrity of that system and significantly alters the probability of a loss-of-coolant accident occurring. Therefore, prompt shutdown of the reactor or isolation of the leaking component is required to reduce the consequences of this event or prevent its occurrence.
- b. The identified leakage rate is restricted to less than 25% of the coolant make-up capability with the minimum charging capacity powered by emergency power. This does allow for further degradation of the system during the evaluation and shutdown process with assurance that adequate cooling make-up capability exists. If the maximum allowable coolant activity existed, the 10 gpm leak rate would not result in doses in excess of the annual average allowed by 10 CFR Part 20.

Should a postulated transient or accident occur (such as a rod ejection or steam line break accident), then, if the primary to secondary leak rate is limited to 0.1 gpm per steam generator, the site boundary dose would be maintained well within the guidelines and all steam generator tubes would maintain their integrity. Continuous operability of two systems of diverse principles is desired to assure some surveillance of coolant leakage. However,

due to the redundancy of systems designed to monitor degradation of the reactor coolant pressure boundary, provisions for short term degradation of one system or long term substitution of a system do not materially alter the degree of safety.

Reference:

- (1) FSAR Section 11.2.3, 14.2.4

3.1.6 Maximum Reactor Coolant Oxygen, Fluoride, and Chloride Concentration

3.1.6.1 With an RCS temperature above 200°F, the RCS chemistry shall be maintained within the following limits.

<u>Contaminant</u>	<u>Steady-State Limit (ppm)</u>	<u>Transient Limit (ppm)</u>
*Oxygen	0.10	1.00
Chloride	0.15	1.50
Fluoride	0.15	1.50

3.1.6.2 With any one or more of the chemistry parameters in excess of its Steady State Limit, but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least hot shutdown within 6 hours and below an RCS temperature of 200°F within the following 30 hours.

3.1.6.3 With any one or more of the chemistry parameters in excess of its Transient Limit, be in at least hot shutdown within 6 hours and below an RCS temperature of 200°F within the following 30 hours and perform an engineering evaluation in accordance with 3.1.6.5.

3.1.6.4 With the RCS temperature at or below 200°F, the RCS chemistry shall be maintained within the following limits.

<u>Contaminant</u>	<u>Normal Limit (ppm)</u>	<u>Transient Limit (ppm)</u>
Oxygen	Saturated	Saturated

* Limits for Oxygen not applicable below 250°F.

11.1

Chloride	0.15	1.50
Fluoride	0.15	1.50

3.1.6.5 If the concentration of chloride or fluoride exceeds the Steady State Limit for more than 48 hours, or exceeds the Transient Limit, maintain the RCS pressure less than 500 psig and perform an engineering evaluation of the effects of the out of limit conditions on the structural integrity of the RCS. This evaluation shall determine that the RCS remains acceptable for continued operation prior to increasing RCS temperature and pressure above 200°F and 500 psig respectively.

Basis:

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the normal limits as specified, the integrity of the Reactor Coolant System is assured under all operating conditions (1). If normal limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin, the addition of hydrazine during subcritical operation, or adjustment of the hydrogen concentration in the volume control tank (2) during power operation. Because of the time dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the period of 24 hours for corrective action to restore concentrations within the limits has been established. If the corrective action has not been effective at the end of the

24 hour period, then the RCS will be brought below 200°F and the corrective action will continue. The effects of contaminants in the reactor coolant are temperature dependent. It is consistent, therefore, to permit a steady state concentration in excess of limit to exist for a longer period of time at the colder RCS temperatures and still provide the assurance that the integrity of the primary coolant system will be maintained. In order to restore the contaminant concentrations to within specification limits in the event such limits were exceeded, mixing of the primary coolant with the reactor coolant pumps may be required. This will result in a small heatup of short duration and will not increase the average coolant temperature above 250°F.

Reference:

- (1) FSAR, Section 4.2
- (2) FSAR, Section 9.2

See Chapter 3.1

3.2.4 With only one of the required boron injection flow paths to the RCS operable, restore at least two boron injection flow paths to the RCS to operable status within 72 hours, or within the next 6 hours be in at least hot shutdown and borated to a shutdown margin equivalent to at least 2.45% delta k/k at cold, no xenon conditions. If the requirements of 3.2.2 are not satisfied within an additional 7 days, then be in cold shutdown within the next 30 hours.

3.2.5 Whenever the RCS temperature is greater than 200°F and is being cooled by the RHR system and the over-pressure protection system is not operable, at least one charging pump shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull-stop position.

LCO 3.4.12

12.i

(4) Refueling water storage tank via gravity feed through manual bypass valve 358 to the suction of the charging pumps.

Available flow paths from the charging pumps to the reactor coolant system include the following:

- (1) Charging flow path through AOV 392A to the RCS Loop B hot leg.
- (2) Charging flow path through AOV 294 to the RCS Loop B cold leg.
- (3) Seal injection flow path to the reactor coolant pumps.

See Chapter 3.12

The rate of boric acid injection must be sufficient to offset the maximum addition of positive reactivity from the decay of xenon after a trip from full power. This can be accomplished through the operation of one charging pump at minimum speed with suction from the refueling water storage tank. Also the time required for boric acid injection allows for the local alignment of manual valves to provide the necessary flow paths.

The quantity of boric acid specified in Table 3.2-1 for each concentration is sufficient at any time in core life to borate the reactor coolant to the required cold shutdown concentration and provide makeup to maintain RCS inventory during the cooldown. The temperature limits specified on Table 3.2-1 are required to maintain solution solubility at the upper concentration in each range. The temperatures listed on Table 3.2-1 are taken from Reference (4). An arbitrary 5°F is added to the Reference (4) for margin. Heat tracing may be used to maintain solution temperature at or above the Table 3.2-1 limits. If the solution temperature of either the flow path or the borated water source is not maintained at or above the minimum temperature specified, the affected flow path must be declared inoperable and the appropriate actions specified in 3.2.4 followed.

~~Placing a charging pump in pull-stop whenever the reactor coolant system temperature is $\geq 200^\circ\text{F}$ and is being cooled by RHR without the overpressure protection system operable will prevent inadvertent overpressurization of the RHR system should letdown be terminated. (3)~~

See Chapter 3.1

References:

- (1) UFSAR Section 9.3.4.2
- (2) RG&E Design Analysis DA-NS-92-133-00 "BAST Boron Concentration Reduction Technical Specification Values" dated Dec. 14, 1992
- (3) L.D. White, Jr. letter A. Schwencer, NRC, Subject: Reactor Vessel Overpressurization, dated February 24, 1977

- d. At or above an RCS temperature of 350°F, two residual heat removal pumps are operable.
- e. At or above an RCS temperature of 350°F, two residual heat removal heat exchangers are operable.
- f. At the conditions required in a through e above, all valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.
- g. At or above an RCS temperature of 350°F, A.C. power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. D.C. control power shall be removed from refueling water storage tank delivery valves 896A, 896B and 856 with the valves open.

See
Chapter
3.5

- h. At or above an RCS temperature of 350°F, check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.2.1 are still applicable.

LC0 3.4.14

13.ix

13.x

- i. Above a reactor coolant system pressure of 1600 psig, except during performance of RCS hydro test, A.C. power shall be removed from accumulator isolation valves 841 and 865 with the valves open.
- j. At or above an RCS temperature of 350° F, A.C. power shall be removed from Safety Injection suction valves 825A and B with the valves in the open position, and from valves 826A, B, C, D with the valves in the closed position.

See
Chapter
3.5

3.3.1.2 If the conditions of 3.3.1.1a are not met, then satisfy the condition within 1 hour or be at hot shutdown in the next 6 hours and at least cold shutdown within an additional 30 hours.

3.3.1.3 The requirements of 3.3.1.1b and 3.3.1.1i may be modified to allow one accumulator to be inoperable or isolated for up to one hour. If the accumulator is not operable or is still isolated after one hour, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS pressure of 1600 psig within an additional 6 hours.

See Chapter 3.5

3.3.1.4 The requirements of 3.3.1.1c may be modified to allow one safety injection pump to be inoperable for up to 72 hours. If the pump is not operable after 72 hours, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS temperature less than 350°F within an additional 6 hours.

3.3.1.5 The requirements of 3.3.1.1d through h. may be modified to allow components to be inoperable at any one time. More than one component may be inoperable at any one time provided that one train of the ECCS is operable. If the requirements of 3.3.1.1d through h. are not satisfied within the time period specified below, the reactor shall be placed in hot shutdown within 6 hours and at an RCS temperature less than 350°F in an additional 6 hours.

LCW 3.4.14

13.x

a. One residual heat removal pump may be out of service provided the pump is restored to operable status within 72 hours.

See Chapter 3.5

- b. One residual heat removal heat exchanger may be out of service for a period of no more than 72 hours.
- c. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours (except as specified in e. below).
- d. Power may be restored to any valve referenced in 3.3.1.1g for the purposes of valve testing provided no more than one such valve has power restored and provided testing is completed and power removed within 12 hours.
- e. Those check valves specified in 3.3.1.1h may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed within 12 hours.

See
Chapter
3.5

LC0 3.4.14

13.xi

~~3.3.1.6 Deleted~~



3.3.1.7

Except during diesel generator load and safeguard sequence testing or when the vessel head is removed, or the steam generator primary system manway is open, no more than one safety injection pump shall be operable whenever the overpressurization protection is provided by a RCS vent of ≥ 1.1 sq. in. (3.15.1.b).

CCO 3.4.12

13.xii

3.3.1.7.1

Whenever only one safety injection pump may be operable by 3.3.1.7, at least two of the three safety injection pumps shall be verified inoperable, as defined in the Basis for this section, a minimum of once per twelve hours.

SR 3.4.12.1

13.xiii

3.3.1.8

Except during diesel generator load and safeguard sequence testing or when the vessel head is removed, or the steam generator primary system manway is open, all three safety injection pumps shall be inoperable and safety injection discharge paths to the RCS isolated whenever overpressure protection is provided by the pressurizer PORVs (3.15.1.a).

CCO 3.4.12

13.xi

3.3.1.8.1

Whenever safety injection pumps are required to be inoperable by 3.3.1.8, the safety injection pumps shall be verified inoperable, as defined in the Basis of this section, a minimum of once per twelve hours. Similarly safety injection discharge paths to the RCS shall be verified to be isolated a minimum of once per twelve hours.

SR 3.4.12.2

13.xiii



3.3.1.8.2

The requirements of 3.3.1.8 may be modified to allow operation of one SI pump provided the associated paths to the RCS are isolated by A.C. power being removed to the discharge MOVs in the closed position, or the manual isolation valves closed. Isolation of the discharge paths shall be verified at least once per 12 hours.

SR 3.4.12.2

13.xiv

See
Chapter
3.7

3.3.5 Control Room Emergency Air Treatment System

3.3.5.1 The RCS temperature shall not be at or above 350°F unless the control room emergency air treatment system is operable.

3.3.5.2 The requirements of 3.3.5.1 may be modified to allow the control room emergency air treatment system to be inoperable for a period of 48 hours. If the system is not made operable within those 48 hours, the reactor shall be placed in hot shutdown within the next 6 hours and the RCS temperature less than 350°F in an additional 12 hours.

Basis

~~The normal procedure for starting the reactor is, first to heat the~~

reactor coolant to near operating temperature by running the Reactor Coolant Pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant. ⁽¹⁾ With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safeguards and auxiliary cooling systems, with the one exception of one fan cooler, as discussed below, are required to be fully operable. During low temperature physics tests, there is a negligible amount of stored energy in the reactor coolant, therefore an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguards systems are not required. The operable status of the various systems and components is to be demonstrated by periodic tests in the Specifications. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will

operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (a) the inoperable component is not repaired within the specified allowable time period or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures will require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. Furthermore, the specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on:

- (1) Assuring with high reliability that the safeguard system will function properly if required to do so.

See
Chapter
2.5

3.15 Overpressure Protection System

Applicability

LCO 3.4.12

25.vi

Applies whenever the temperature of one or more of the RCS cold legs is $\leq 330^\circ\text{F}$, or the Residual Heat Removal System is in operation.

Objective

To prevent overpressurization of the reactor coolant system and the residual heat removal system.

Specification

3.15.1

LCO 3.4.12

25.i

25.ii

25.iii

Except during secondary side hydrostatic tests in which RCS pressure is to be raised above the PORV setpoint, at least one of the following over-pressure protection systems shall be operable:

- a. Two pressurizer power operated relief valves (PORVs) with a lift setting of ≤ 424 psig, or
- b. A reactor coolant system vent of ≥ 1.1 square inches:

3.15.1.1

LCO 3.4.12

25.v

With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the RCS through a 1.1 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.2

LCO 3.4.12

With both PORVs inoperable, depressurize and vent the RCS through a 1.1 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.3

25.v

~~Use of the overpressure protection system to mitigate an RCS or RHRS pressure transient shall be reported in accordance with 6.9.2.~~

Basis

~~At RCS vent opening of greater than 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq 330^\circ\text{F}$ ⁽¹⁾. This relief capacity will~~

ensure that no overpressurization of the RHR system could occur. The vent opening protects the RCS and RHRS from overpressurization when the transient is limited to either 1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq 50^{\circ}\text{F}$ above the RCS cold leg temperature or 2) the start of a safety injection pump and its injection into a water solid RCS⁽²⁾.

The operability of two pressurizer PORVs ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq 330^{\circ}\text{F}$ ⁽²⁾. This relief capacity will also ensure that no overpressurization of the RHR system could occur. Either PORV has adequate relieving capability to protect the RCS and RHRS from overpressurization when the transient is limited to either 1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq 50^{\circ}\text{F}$ above the RCS cold leg temperature or 2) charging/letdown mismatch with three charging pumps in operation⁽³⁾.

References:

- (1) L. D. White, Jr., letter to A. Schwencer, NRC, dated July 29, 1977.
- (2) SER for SEP Topics V-10.B, V-11.B, VII-3, "Safe Shutdown," dated September 29, 1981.
- (3) Westinghouse Report, "R. E. Ginna Low Temperature Overpressure Protection System (LTOPS) Setpoint Phase II Evaluation Final Report," dated February 1991 submitted by letter to Allen R. Johnson, NRC, dated February 15, 1991.

SR 3.4.2.2
 SR 3.4.2.1
 SR 3.4.3.1
 SR 3.4.1.1
 SR 3.4.1.2
 SR 3.4.1.3

28.i.b

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

See Chapter 3.3

<u>Channel Description</u>	<u>Check</u>	<u>Calibrate</u>	<u>Test</u>	<u>Remarks</u>
1. Nuclear Power Range	S M*(3)	D(1) Q*(3)	B/W(2)(4) P(2)(5)	1) Heat balance calculation** 2) Signal to ΔT ; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset** 4) High setpoint ($<109\%$ of rated power) 5) Low setpoint ($<25\%$ of rated power)
2. Nuclear Intermediate Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3. Nuclear Source Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4. Reactor Coolant Temperature	S	R	M(1) (2)	1) Overtemperature-Delta T 2) Overpower - Delta T
5. Reactor Coolant Flow	S	R	M	
6. Pressurizer Water Level	S	R	M	
7. Pressurizer Pressure	S	R	M	
8. 4 Kv Voltage & Frequency	N.A.	R	M	Reactor Protection circuits only
9. Rod Position Indication	S(1,2)	N.A.	M	1) With step counters 2) Log rod position indications each 4 hours when rod deviation monitor is out of service

* By means of the movable in-core detector system.

** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

TABLE 4.1-1 (Continued)

<u>Channel Description</u>	<u>Check</u>	<u>Calibrate</u>	<u>Test</u>	<u>Remarks</u>
10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With rod position indication 2) Log rod position indications each 4 hours when rod deviation monitor is out of service
11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Storage Tank Level	D	R	N.A.	Note 4
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
19. Boric Acid Control	N.A.	R	N.A.	
SR 3.4.15.320. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
24. Accumulator Level and Pressure	S	R	N.A.	

Amendment No. ~~22~~ 57

4.1-6

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	<u>Test</u>	<u>Frequency</u>
1. Reactor Coolant Chemistry Samples	Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2. Reactor Coolant Boron	Boron Concentration	Weekly
3. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly
4. Boric Acid Storage Tank	Boron Concentration	Twice/Week ⁽⁴⁾
5. Control Rods	Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (1)
6a. Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system	Monthly
6b. Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur	Each Refueling Shutdown
7. Pressurizer Safety Valves	Set point	Each Refueling Shutdown
8. Main Steam Safety Valves	Set point	Each Refueling Shutdown
9. Containment Isolation Trip	Functioning	Each Refueling Shutdown
10. Refueling System Interlocks	Functioning	Prior to Refueling Operations

See Chapters
3.6, 3.7, and 3.9

See Chapters 3.5
3.6, 3.7, and 5.0

	<u>Test</u>	<u>Frequency</u>
11. Service Water System	Functioning	Each Refueling Shutdown
12. Fire Protection Pump and Power Supply	Functioning	Monthly
13. Spray Additive Tank	NaOH Concent	Monthly
14. Accumulator	Boron Concentration	Bi-Monthly
15. Primary System Leakage	Evaluate	Daily
16. Diesel Fuel Supply	Fuel Inventory	Daily
17. Spent Fuel Pit	Boron Concentration	Monthly
18. Secondary Coolant Samples	Gross Activity	72 hours (2) (3)
19. Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown

See 3.4.3.1
28.ii.f

See Chapters 3.3, 3.5 and 3.9

Notes:

- (1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.
- (2) Not required during a cold or refueling shutdown.
- (3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.
- (4) When BAST is required to be operable.

TABLE 4.1-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE
AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS, FREQUENCY</u>	<u>MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED</u>
SR 3.4.16.1 <u>2P.IV.a</u> 1. Gross Activity Determination (beta-gamma) (1)	At least once per 72 hours	Above cold shutdown
SR 3.4.16.2 <u>2P.IV.b</u> 2. Isotopic Analysis for Dose Equivalent I-131 Concentra- tion	1 per 14 days	Above 5% reactor power
SR 3.4.16.3 <u>2P.IV.c</u> 3. Radiochemical for E Determination (2)	1 per 6 months (3)	Above 5% reactor power
SR 3.4.16.2 4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	LCO 3.4.16 a) Once per 8 hours, whenever the I-131 equivalent activity exceeds the limit of 3.1.4.1.b LCO 3.4.16 b) One sample between 2 and .10 hours following a reactor power change exceeding 15 per- cent within a 1-hour period	As required by Specification 3.1.4.3.c* Hot shutdown or above

(1) A gross radiactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units $\mu\text{Ci/gm}$. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of E.

(3) Sample to be taken after a minimum of 2 EFPD and 31 ~~20~~ days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.

* Except at refueling shutdown, sampling shall be continued until the activity of the reactor coolant system is restored to within its limits.

See Chapter 5.0

Table 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

	<u>Instrument</u>	<u>Channel Check</u>	<u>Source Check</u>	<u>Functional Test</u>	<u>Channel Calibration</u>
1.	Gross Activity Monitor (Liquid)				
a.	Liquid Rad Waste (R-18)	D(7)	M(4)	Q(1)	R(5)
b.	Steam Generator Blowdown (R-19)	D(7)	M(4)	Q(1)	R(5)
c.	Turbine Building Floor Drains (R-21)	D(7)	M(4)	Q(1)	R(5)
d.	High Conductivity Waste (R-22)	D(7)	M(4)	Q(1)	R(5)
e.	Containment Fan Coolers (R-16)	D(7)	M(4)	Q(2)	R(5)
f.	Spent Fuel Pool Heat Exchanger A Loop (R-20A)	D(7)	M(4)	Q(2)	R(5)
g.	Spent Fuel Pool Heat Exchanger B Loop (R-20B)	D(7)	M(4)	Q(2)	R(5)
	Plant Ventilation				
a.	Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D(7)	M	Q(1)	R(5)
b.	Particulate Sampler (R-13)	W(7)	N.A.	N.A.	R(5)
c.	Iodine Sampler (R-10B and R-14A)	W(7)	N.A.	M	R(5)
d.	Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3.	Containment Purge				
SR 3.4.15.1, 2, 4 a.	Noble Gas Activity (R-12)	D(7)	PR	Q(1)	R(5)
SR 3.4.15.1, 2, 4 b.	Particulate Sampler (R-11)	W(7)	N.A.	Q(1)	R(5)
	c. Iodine Sampler (R-10A and R-12A)	W(7)	N.A.	M	R(5)
	d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
	Air Ejector Monitor (R-15 and R-15A)	D(7)	M	M(2)	R(5)
5.	Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)
6.	Main Steam Lines (R-31 and R-32)	M	N.A.	Q	R

See also Chapter 3.3

See Chapter 5.0

4.2.1.1 The inspection interval for Quality Group A components shall be ten year intervals of service commencing on January 1, 1970.

4.2.1.2 The inspection intervals for Quality Group B and C Components shall be ten year intervals of service commencing with May 1, 1973, January 1, 1980, 1990 and 2000, respectively.

4.2.1.3 The inspection intervals for the High Energy Piping Outside of Containment shall be ten year intervals of service commencing May 1, 1973, January 1, 1980, 1990 and 2000, respectively. The inspection program during each third of the first inspection interval provides for examination of all welds at design basis break locations and one-third of all welds at locations where a weld failure would result in unacceptable consequences. During each succeeding inspection interval, the program shall provide for an examination of each of the design basis break location welds, and each of the welds at locations where a weld failure would result in unacceptable consequences.

4.2.1.4 The inspection intervals for Steam Generator Tubes shall be specified in the "Inservice Inspection Program" for the applicable forty month period commencing with May 1, 1973.

4.2.1.4.a Steam generator tubes that have imperfections greater than 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.

4.2.1.4.b Steam generator sleeves that have imperfections greater than 30% through wall, as indicated by eddy current, shall be repaired by plugging.

~~Amendment No. 8, 35, 37 4.2-2~~

~~Correction letter of July 3, 1990~~

See Chapter 5.02

SR 3.4.15.2

See Chapter 5.0

- SR 3.4.4.3
- SR 3.4.7.3
- SR 3.4.8.2
- SR 3.4.9.2
- SR 3.4.11.2

30.iii

4.3 Reactor Coolant System

Applicability

Applies to surveillance of the reactor coolant system and its components.

Objective

To ensure operability of the reactor coolant system and its components.

Specifications:

4.3.1 Reactor Vessel Material Surveillance Testing

30.vii

4.3.1.1 ~~The reactor vessel material surveillance specimens shall be removed and examined to determine changes in their material properties, as required by Appendix H to 10 CFR Part 50.~~

4.3.2 Pressurizer

4.3.2.1 The pressurizer water level shall be verified to be within its limits at least once per 12 hours during power operation and hot shutdown.

SR 3.4.9.1

4.3.3 Check Valves

4.3.3.1 Leakage testing of check valves 853A, 853B, 867A, 867B, 878G and 878J shall be accomplished prior to criticality, except for low power physics testing, following (1) refueling, ~~(2) cold shutdown,~~ and (3) maintenance, repair or replacement work on the valves. Leakage may be measured indirectly from the performance of pressure indicators, system volume measurements or by direct measurement. Minimum test differential pressure shall be greater than 150 psid. See 4.3.3.4 for allowable leakage rates.

SR 3.4.14.1

30.iv

30.viii

4.3.3.2 Check valves 878G and 878J will be tested for leakage following each safety injection flow test. Minimum test differential pressure shall be greater than 150 psid. See 4.3.3.4 for allowable leakage rates.

SR 3.4.14.1

30.iv

4.3.3.3 Motor-operated valves 878A and 878C and check valves 877A, 877B, 878F, and 878H shall be tested at the first refueling outage following the date of this order* to individually assure integrity of at least two of the three pressure boundaries in each hot leg high-head safety injection path. Testing shall also be performed after any opening of either motor-operated valve and at a minimum, once every 40 months. Opening of the motor-operated valves, and testing, are to be performed at a test pressure less than that of the

SR 3.4.14.2

30.v

lowest design pressure of any portion of the high-head safety injection system which may be pressurized during the test. Minimum test differential pressure shall be greater than 150 psid. See 4.3.3.4 for allowable leakage rates.

4.3.3.4 Allowable check valve leakage rates are as follows:

SR 3.4.14.1

30.v

- (a) Leakage rates less than or equal to 1.0 gpm are considered acceptable.
- (b) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
- (c) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
- (d) Leakage rates greater than 5.0 gpm are considered unacceptable.

4.3.4 Relief Valves

4.3.4.1 Each PORV shall be demonstrated operable at least once per 18 months by performance of a CHANNEL CALIBRATION.

SE 3.4.4.2.6

4.3.4.2 Except during cold and refueling shutdown each block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel unless the valve is already closed.

SR 3.4.4.1

4.3.5 Reactor Coolant Loops

30. ix

due to PORV leakage > 10 gpm

4.3.5.1 When reactor power is above 130 MWt (8.5%), the reactor coolant pumps shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

SR 3.4.4.1

4.3.5.2 When the average coolant temperature is above 350°F but the reactor is not critical, when the reactor is at hot shutdown, or when the reactor is critical but reactor power is less than or equal to 130 MWt (8.5%):

SE 3.4.5.1

a) the operating reactor coolant pump(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours, and

SR 3.4.5.2

b) if a reactor coolant pump is not operating, but must be operable, it shall be demonstrated operable once per 7 days by verifying correct breaker alignments and indicated power availability.

4.3.5.3 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F, and fuel is in the reactor, the following shall be performed to demonstrate a loop is operable. Tests need not be performed if a loop is not relied upon to satisfy the requirements of Specification 3.1.1.1.e.

SR 3.4.1.2 a) to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.

30.ii

~~b) to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.~~

SR 3.4.5.1
SR 3.4.5.2
SR 3.4.8.1
4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor, at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

SR 3.4.5.2
SR 3.4.5.2
SR 3.4.7.2
4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable, the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

30.i

~~4.3.5.6 Each reactor coolant system vent path shall be demonstrated operable at least once per 18 months by:~~

- ~~1. Verifying all manual isolation valves in each vent path are locked in the open position.~~
- ~~2. Verifying flow through the reactor coolant vent system vent paths using either liquid or gas.~~

Basis:

~~This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to~~

neutron irradiation and the thermal environment. The test data obtained from this program will be used to determine the conditions under which the reactor vessel can be operated with adequate margins of safety against fracture throughout its service life.

The surveillance requirements on pressurizer equipment will assure proper performance of the pressurizer function and give early indication of malfunctions.

4.16 Overpressure Protection System

Applicability:

Applies to the reactor coolant system overpressure protection system.

43.i

Objective:

To verify that the overpressure protection system will function properly if needed.

Specification

4.16.1 Each PORV shall be demonstrated operable by:

SR 3.4.12.5

43.ii

a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

SR 3.4.12.8

b. Performance of a channel calibration on the PORV actuation channel at least once per 18 months.

SR 3.4.12.4

c. Verifying the PORV isolation valve is open at least once per 72 hours when the overpressure protection system is required to be operable.

4.16.2 The RCS vent(s) shall be verified to be open at least once per 12 hours when the vent(s) is being used for overpressure protection except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position. Then verify these valves open at least once per 31 days.

SR 3.4.12.3

SR 3.4.12.3

SR 3.4.12.7

} 43.i

LCO 3.5.2 (13.v)

3.3

Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, and Charcoal/HEPA Filters

Objective

To define those conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident, and (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

3.3.1 Safety Injection and Residual Heat Removal Systems

3.3.1.1 The reactor shall not be taken above the mode indicated unless the following conditions are met:

- a. Above cold shutdown, the refueling water storage tank contains not less than 300,000 gallons of water, with a boron concentration of at least ~~2000~~ ²³⁰⁰ ppm, and ~~≤ 2000 ppm~~ ^{≤ 2300 ppm}.
- b. Above a reactor coolant system pressure of 1600 psig, except during performance of RCS hydro test, each accumulator is pressurized to at least 700 psig with an indicated level of at least 50% and a maximum of 82% with a boron concentration of at least ~~1800~~ ²¹⁰⁰ ppm, and ~~≤ 2100 ppm~~ ^{≤ 2400 ppm}.
- c. At or above a reactor coolant system temperature of 350°F, three safety injection pumps are operable.

(13.ii)
LCO 3.5.4
SR 3.5.4.1
SR 3.5.4.2

LCO 3.5.1
(13.i)
(13.vi)
(13.vii)

LCO 3.5.2
(13.iii)

LCO 3.5.2 d. At or above an RCS temperature of 350°F, two residual heat removal pumps are operable.

LCO 3.5.2 e. At or above an RCS temperature of 350°F, two residual heat removal heat exchangers are operable.

LCO 3.5.1
LCO 3.5.2
LCO 3.5.4 f. At the conditions required in a through e above, all valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

LCO 3.5.2 g. At or above an RCS temperature of 350°F, A.C. power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. D.C. control power shall be removed from refueling water storage tank delivery valves 896A, 896B and 856 with the valves open.

13.viii

See Chapter 3.4

h. At or above an RCS temperature of 350°F, check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.2.1 are still applicable.

LCO 3.5.1
LCO 3.5.15 i. Above a reactor coolant system pressure of 1600 psig, except during performance of RCS hydro test, A.C. power shall be removed from accumulator isolation valves 841 and 865 with the valves open.

LCO 3.5.2 j. At or above an RCS temperature of 350° F, A.C. power shall be removed from Safety Injection suction valves 825A and B with the valves in the open position, and from valves 826A, B, C, D with the valves in the closed position.

3.3.1.2 If the conditions of 3.3.1.1a are not met, then satisfy the condition within 1 hour or be at hot shutdown in the next 6 hours and at least cold shutdown within an additional 30 hours.

LCO 3.5.4

13.2

3.3.1.3 The requirements of 3.3.1.1b and 3.3.1.1i may be modified to allow one accumulator to be inoperable or isolated for up to one hour. If the accumulator is not operable or is still isolated after one hour, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS pressure of 1600 psig within an additional 6 hours.

LCO 3.5.1

13.4

3.3.1.4 The requirements of 3.3.1.1c may be modified to allow one safety injection pump to be inoperable for up to 72 hours. If the pump is not operable after 72 hours, the reactor shall be placed in hot shutdown within the following 6 hours and below a RCS temperature less than 350°F within an additional 6 hours.

LCO 3.5.2

3.3.1.5 The requirements of 3.3.1.1d through h. may be modified to allow components to be inoperable at any one time. More than one component may be inoperable at any one time provided that one train of the ECCS is operable. If the requirements of 3.3.1.1d through h. are not satisfied within the time period specified below, the reactor shall be placed in hot shutdown within 6 hours and at an RCS temperature less than 350°F in an additional 6 hours.

LCO 3.5.2

- a. One residual heat removal pump may be out of service provided the pump is restored to operable status within 72 hours.

LCO 3.5.2

LCO 3.5.2 b. One residual heat removal heat exchanger may be out of service for a period of no more than 72 hours.

LCO 3.5.2 c. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours (except as specified in e. below).

LCO 3.5.2 (Note) 13.iv d. Power may be restored to any valve referenced in 3.3.1.1g for the purposes of valve testing provided no more than one such valve has power restored and provided testing is completed and power removed within 12 hours.

~~e. Those check valves specified in 3.3.1.1h may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed within 12 hours.~~

~~3.3.1.6 Deleted~~

operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. If it develops that (a) the inoperable component is not repaired within the specified allowable time period or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures will require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. Furthermore, the specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on:

- (1) Assuring with high reliability that the safeguard system will function properly if required to do so.

(2) Allowances of sufficient time to effect repairs using safe and proper procedures.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.

<u>Time After Shutdown</u>	<u>Decay Heat % of Rated Power</u>
1 min.	4.5
30 min.	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition, significantly reduces the potential consequences of a loss-of-coolant accident; and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and therefore in such a case the reactor is to be put into the cold shutdown condition.

See Chapter 3.4

With respect to the core cooling function, there is some functional re-

See Chapter
3.7

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident.⁽⁸⁾ The control room emergency air treatment system is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the Design Basis Accident.⁽⁹⁾ Reactor operation may continue for a limited time while repairs are being made to the air treatment system since it is unlikely that the system would be needed. Technical Specification 3.3.5 applies only to the equipment necessary to filter the control room atmosphere. Equipment necessary to initiate isolation of the control room is covered by another specification.

~~The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet.~~

~~The limitation of no more than one safety injection pump to be operable when overpressure protection is being provided by a RCS vent of ≥ 1.1 sq. in. insures~~

TABLE 4.1-1 (Continued)

See Chapter 3.3

SR 3.5.4.1

See Chapter 3.3

SR 3.5.2.5
SR 3.5.2.6

SR 3.5.1.2
SR 3.5.1.3
28.6.5

See Chapter 3.3

Channel Description	Check	Calibrate	Test	Remarks
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10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With rod position indication 2) Log rod position indications each 4 hours when rod deviation monitor is out of service
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11. Steam Generator Level	S	R	M	
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12. Charging Flow	N.A.	R	N.A.	
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13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
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14. Boric Acid Storage Tank Level	D	R	N.A.	Note 4
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15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
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16. Volume Control Tank Level	N.A.	R	N.A.	
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17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
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18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
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19. Boric Acid Control	N.A.	R	N.A.	
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20. Containment Drain Sump Level	N.A.	R	N.A.	
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21. Valve Temperature Interlocks	N.A.	N.A.	R	
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22. Pump-Valve Interlock	R	N.A.	N.A.	
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23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
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24. Accumulator Level and Pressure	S	R	N.A.	
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Added:

SR 3.5.2.3
 SR 3.5.4.2
 SR 3.5.1.5
 SR 3.5.2.7
 SR 3.5.1.1
 SR 3.5.1.3
 SR 3.5.1.4

28.ii.i

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	<u>Test</u>	<u>Frequency</u>
1. Reactor Coolant Chemistry Samples	Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2. Reactor Coolant Boron	Boron Concentration	Weekly
3. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly
4. Boric Acid Storage Tank	Boron Concentration	Twice/Week ^(a)
5. Control Rods	Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (1)
6a. Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system	Monthly
6b. Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur	Each Refueling Shutdown
7. Pressurizer Safety Valves	Set point	Each Refueling Shutdown
8. Main Steam Safety Valves	Set point	Each Refueling Shutdown
9. Containment Isolation Trip	Functioning	Each Refueling Shutdown
10. Refueling System Interlocks	Functioning	Prior to Refueling Operations

See chapters 3.3,
3.1, 3.7 and 3.9

See
chapter
3.4

SR 3.5.4.2

28.ii.i

See chapters 3.5,
3.6, 3.7, and 5.0

	<u>Test</u>	<u>Frequency</u>
11. Service Water System	Functioning	Each Refueling Shutdown
12. Fire Protection Pump and Power Supply	Functioning	Monthly
13. Spray Additive Tank	NaOH Concent	Monthly
SR 3.5.1.4 14. Accumulator	Boron Concentration	Bi-Monthly 28.11.1
15. Primary System Leakage	Evaluate	Daily
16. Diesel Fuel Supply	Fuel Inventory	Daily
17. Spent Fuel Pit	Boron Concentration	Monthly
18. Secondary Coolant Samples	Gross Activity	72 hours (2) (3)
19. Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown

SR 3.5.1.4

See
Chapters
3.3, 3.8,
and 3.9

Notes:

- (1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.
- (2) Not required during a cold or refueling shutdown.
- (3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.
- (4) When BAST is required to be operable.

4.3.4 Relief Valves

4.3.4.1 Each PORV shall be demonstrated operable at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.3.4.2 Except during cold and refueling shutdown each block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel unless the valve is already closed.

4.3.5 Reactor Coolant Loops

4.3.5.1 When reactor power is above 130 Mwt (8.5%), the reactor coolant pumps shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.2 When the average coolant temperature is above 350°F but the reactor is not critical, when the reactor is at hot shutdown, or when the reactor is critical but reactor power is less than or equal to 130 Mwt (8.5%):

- a) the operating reactor coolant pump(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours, and
- b) if a reactor coolant pump is not operating, but must be operable, it shall be demonstrated operable once per 7 days by verifying correct breaker alignments and indicated power availability.

See
Chapter
3.4

4.3.5.3 When the ~~reactor is at cold shutdown or when the~~ average coolant temperature is between 200°F and 350°F, and fuel is in the reactor, the following shall be performed to demonstrate a loop is operable. Tests need not be performed if a loop is not relied upon to satisfy the requirements of Specification 3.1.1.1.e.

See Chapter
3.4

a) ... to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.

b) to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.

SR 3.5.3.1

See
Chapter
3.4

4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor, at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable, the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

4.3.5.6 Each reactor coolant system vent path shall be demonstrated operable at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Verifying flow through the reactor coolant vent system vent paths using either liquid or gas.

Basis:

This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to

SR 3.5.2.1 }
SR 3.5.2.2 } (32.iv)

4.5 Safety Injection, Containment Spray and Iodine Removal

Systems Tests

Applicability:

Applies to testing of the Safety Injection System, the Containment Spray System, and the Air Filtration System inside Containment.

Objective:

To verify that the subject systems will respond promptly and perform their intended functions, if required.

Specification:

4.5.1 Safety Tests

4.5.1.1 Safety Injection System

- a. System tests shall be performed at each reactor refueling interval. The test shall be performed in accordance with the following:

SR 3.5.2.5

(32.i)

With the reactor coolant system pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will

be applied to initiate operation of the system. The safety injection and residual heat removal pump motors are prevented from starting during the test.

SR 3.5.2.5

b. The system test will be considered satisfactory if control board indication and visual observations indicate that all valves have received the Safety Injection signal and have completed their travel. The proper sequence and timing of the rotating components are to be verified in conjunction with Section 4.6.1 b.

See Chapter 3.6

4.5.1.2 Containment Spray System

- a. System tests shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves, in the spray supply lines, at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.
- b. The spray nozzles shall be checked for proper functioning at least every five years.
- c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.5.2 Component Tests

4.5.2.1 Pumps

- a. Except during cold or refueling shutdowns the safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started at intervals not to exceed one month. The pumps shall be tested prior to startup if the time since the last test exceeds 1 month.

SR 3.5.3.1
SR 3.5.2.4

32.ii

SR 3.5.3.1
SR 3.5.2.4
SR 3.5.2.3
32.22

b. Acceptable levels of performance for the pumps shall be that the pumps start, operate, and develop the minimum discharge pressure for the flows listed in the table below:

PUMPS	RECYCLE FLOW RATE	DISCHARGE PRESSURE
Containment Spray Pumps	35 gpm	240 psig
Residual Heat Removal Pumps	[200 gpm] 450 gpm	[140 psig] 138 psig
Safety Injection Pumps	[50 gpm] 150 gpm	[1420 psig] 1356 psig

Notes

(1)

(2)

Table 4.5-1

Notes

- (1) Items in square brackets are effective until the installation of the new residual heat removal minimum flow recirculation system.
- (2) Items in square brackets are effective until installation of the new safety injection minimum flow recirculation system.

See Chapter 3.6

4.5.2.2 Valves

a. Except during cold or refueling shutdowns the spray additive valves shall be tested at intervals not to exceed one month. With the pumps shut down and the valves upstream and downstream

of the spray additive valves closed, each valve will be opened and closed by operator action. This test shall be performed prior to startup if the time since the last test exceeds one month.

- 3. The accumulator check valves shall be checked for operability during each refueling shutdown.

32.iii

4.5.2.3 Air Filtration System

4.5.2.3.1 At least once every 18 months or after every 720 hours of charcoal filtration system operation since the last test, or following painting, fire or chemical release in any ventilation zone communicating with the system, the post accident charcoal system shall have the following conditions demonstrated.

- a. The pressure drop across the charcoal adsorber bank is less than 3 inches of water at design flow rate ($\pm 10\%$).
- b. In place Freon testing, under ambient conditions, shall show at least 99% removal.
- c. The iodine removal efficiency of at least one charcoal filter cell shall be measured. The filter cell to be tested shall be selected randomly from those cells with the longest in-bank residence time. The minimum acceptable value for filter efficiency is 90% for removal of methyl iodide when tested at at least 286°F and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

See Chapter 5.0

- See Chapter 3.8
- c. The tests in Specification 4.6.1b will be performed prior to exceeding cold shutdown if the time since the last test exceeds 31 days.
 - d. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment.
 - e. At least once per 18 months during shutdown by:
 1. Inspecting the diesel in accordance with the manufacturer's recommendations for this class of standby service.
 2. Verifying the generator capability to reject a load of 295 KW without tripping.
 3. Simulating a loss of offsite power in conjunction with a safety injection test signal and:
 - (a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.
 - (b) Verifying the diesel starts from normal standby condition on the auto-start signal, energizes the automatically connected emergency loads with the following maximum breaker closure times after the initial starting signal for Trains A and B not being exceeded

SR 3.5.2.6

(33.vi)

	A	B
Diesel plus Safety Injection Pump plus RHR Pump	20 sec	22 sec
All Breakers	40 sec	42 sec

and operates for \geq five minutes while its generator is loaded with emergency loads.

- See Chapter 3.8
- (c) Verifying that all diesel generator trips, except engine overspeed, low lube oil pressure, and overcrank, are automatically bypassed upon a safety injection actuation signal.

3.3.2 Containment Cooling and Iodine Removal

3.3.2.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

- 3.6.6
- a. The spray additive tank contains not less than 4500 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.
 - b. Both containment spray pumps are operable.
 - c. Four recirculation fan cooler units including the associated HEPA filter units with demisters are operable.
 - d. Both post accident charcoal filter units are operable.
 - e. All valves and piping associated with the above components which are required to function during accident conditions are operable.

3.3.2.2 The requirements of 3.3.2.1 may be modified to allow components to be inoperable at any one time provided that 1) the time limits and other requirements specified in a through f below are satisfied, and 2) at least 1 containment spray pump, ~~1~~² fan cooler units, ~~1~~² HEPA filter units with demisters, and 1 charcoal filter unit and all required valves and piping associated with these components are operable. If these requirements are not satisfied, the reactor shall be in hot shutdown within 6 hours. If the requirements are not satisfied within an additional 48 hours, be in cold shutdown within the next 30 hours.

- 3.6.6
- (13.xv)
- a. One ^{or 2} HEPA filter unit or demister and/or associated recirculation fan cooler may be inoperable for a period of no more than 7 days.

3.6.6 b. One containment spray pump may be inoperable provided the pump is restored to operable status within 3 days.

3.6.6 c. Any valve or piping in a system, required to function during accident conditions, may be inoperable provided it is restored to operable status within 72 hours.

3.6.6 d. One post accident charcoal filter unit and/or its associated fan cooler may be inoperable provided the unit is restored to operable status within 7 days.

(13.xv)

Two post-accident charcoal filter trains may be inoperable up to 72 hours if both CS trains are available.

3.6.6 e. The spray additive system may be inoperable for a period of no more than 3 days provided that both charcoal filter units are operable.

3.3.3 Component Cooling System

3.3.3.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

- a. Both component cooling pumps are operable.
- b. Both component cooling heat exchangers are operable.
- c. All valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

3.3.3.2 The requirements of 3.3.3.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.3.1 within the time period

Addressed in chapter 3.3

3.3-6

redundancy for certain ranges of break sizes. (2)

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus post accident charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, only two of the four fan-coolers are required to remove heat lost from equipment and piping within containment. (3) In the event of a Design Basis Accident, any one of the following will serve to reduce airborne iodine activity and maintain doses within the values calculated in the FSAR: (1) two containment spray pumps and sodium hydroxide addition, (2) two fan-coolers and two post accident charcoal filters, or (3) one containment spray pump and sodium hydroxide addition plus one fan-cooler and one post accident charcoal filter. (4) In addition, the containment integrity analysis assumes that one containment spray pump and two fan-coolers operate to reduce containment pressure following a Design Basis Accident. (9) Because of the difficulty of access to make repairs to a fan-cooler and because of the low probability of a Design Basis Accident during that time, an additional seven days operation with an inoperable fan-cooler is permitted. The containment spray pumps and spray additive system are located outside containment and are, therefore, less difficult to repair. Therefore, three days with an inoperable containment spray pump or spray additive system is deemed acceptable.

The Component Cooling System is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. (5) In addition, if during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected. (6)(7)

See
Chapter
3.7

References

- (1) Deleted
- (2) UFSAR Section 6.3.3.1
- (3) UFSAR Section 6.2.2.1
- (4) UFSAR Section 15.6.4.3
- (5) UFSAR Section 9.2.2.4
- (6) UFSAR Section 9.2.2.4
- (7) Deleted
- (8) UFSAR Section 9.2.1.2
- (9) UFSAR Section 6.2.1.1 (Containment Integrity) and UFSAR Section 6.4 (CR Emergency Air Treatment)
- (10) Westinghouse Report, "R.E. Ginna Boric Acid Storage Tank Boron Concentration Reduction Study" dated Nov. 1992 by C.J. McHugh and J.J. Spryshak

3.6 Containment System

Applicability:

Applies to the integrity of reactor containment.

Objective:

To define the operating status of the reactor containment for plant operation.

Specification:

3.6.1 Containment Integrity

3.6.1
14.i

a. Except as allowed by 3.6.3, containment integrity shall not be violated unless the reactor is in the cold shutdown condition. Closed valves may be opened on an intermittent basis under administrative control.

Note 1 →

Addressed
in Chapter
3.9

b. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is greater than 2000 ppm.

c. Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact unless the boron concentration is greater than 2000 ppm.

3.6.2 Internal Pressure

3.6.4
16.ii

If the internal pressure exceeds 1 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within ~~24~~ hours or the reactor rendered subcritical.

8

3.6.3 Containment Isolation Boundaries

3.6.3.1 With a containment isolation boundary inoperable for one or more containment penetrations, either:

LCO 3.6.2
LCO 3.6.3

- a. Restore each inoperable boundary to OPERABLE status within 4 hours, or
- b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, one closed manual valve, or a blind flange, or
- c. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

16.iii

16.iv

3.6.4 Combustible Gas Control

3.6.4.1 When the reactor is critical, at least two independent containment hydrogen monitors shall be operable. One of the monitors may be the Post Accident Sampling System.

3.6.4.2 With only one hydrogen monitor operable, restore a second monitor to operable status within 30 days or be in at least hot shutdown within the next 6 hours.

3.6.4.3 With no hydrogen monitors operable, restore at least one monitor to operable status within 72 hours or be in at least hot shutdown within the next 6 hours.

Addressed in Chapter 3.3

3.6.5 Containment Mini-Purge

LCO 3.6.3

16.v

Whenever the containment integrity is required, emphasis will be placed on limiting all purging and venting times to as low as achievable. The mini-purge isolation valves will remain closed to the maximum extent practicable but may be open for pressure control, for ALARA, for respirable air quality considerations for personnel entry, for surveillance tests that may require the valve to be open or other safety related reasons.

Under Admin control

16.vi

- LCO 3.6.5

16.vii

- LCO 3.6.3

referred to base for LCO 3.6.3

Basis:

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the reactor coolant system ruptures.

The shutdown margins are selected based on the type of activities that are being carried out. The (2000 ppm) boron concentration provides shutdown margin which precludes criticality under any circumstances. When the reactor head is not to be removed, a cold shutdown margin of 1%Δk/k precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major steam break accident were as much as 1 psig.⁽¹⁾ The containment is designed to withstand an internal vacuum of 2.5 psig.⁽²⁾ The 2.0 psig vacuum is specified as an operating limit to avoid any difficulties with motor cooling.

In order to minimize containment leakage during a design basis accident involving a significant fission product release, penetrations not required for accident mitigation are provided with isolation boundaries. These isolation boundaries consist of either passive devices or active automatic valves and are listed in a procedure under the control of the Quality Assurance Program. Closed manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges and closed systems are considered passive devices. Automatic isolation valves designed to close following an accident without operator action, are considered active devices. Two isolation devices are provided for each mechanical penetration, such that no single credible failure or malfunction of an active component can cause a loss of isolation, or result in a leakage rate that exceeds limits assumed in the safety analyses⁽³⁾.

In the event that one isolation boundary is inoperable, the affected penetration must be isolated with at least one boundary that is not affected by a single active failure. Isolation boundaries that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve, or a blind flange.

The opening of closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an individual qualified in accordance with station procedures, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to isolate the boundary and that this action will prevent the release of radioactivity outside the containment.

References:

- (1) Westinghouse Analysis, "Report for the BAST Concentration Reduction for R. E. Ginna", August 1985, submitted via Application for Amendment to the Operating License in a letter from R.W. Kober, RG&E to H.A. Denton, NRC, dated October 16, 1985
- (2) UFSAR - Section 6.2.1.4
- (3) UFSAR - Section 6.2.4

CTS page 4.1-7 is no longer contained in Attachment B, Section 3.6.

SR 3.6.5.1 }
SR 3.6.6.8 }

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TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	<u>Test</u>	<u>Frequency</u>
1.	Reactor Coolant Chemistry Samples Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2.	Reactor Coolant Boron	Boron Concentration Weekly
3.	Refueling Water Storage Tank Water Sample	Boron Concentration Weekly
4.	Boric Acid Storage Tank	Boron Concentration Twice/Week ⁽⁴⁾
5.	Control Rods	Rod drop times of all full length rods After vessel head removal and at least once per 18 months (1)
6a.	Full Length Control Rod	Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system Monthly
6b.	Full Length Control Rod	Move each rod through its full length to verify that the rod position indication system transitions occur Each Refueling Shutdown
7.	Pressurizer Safety Valves	Set point Each Refueling Shutdown
8.	Main Steam Safety Valves	Set point Each Refueling Shutdown
9.	Containment Isolation Trip	Functioning Each Refueling Shutdown
10.	Refueling System Interlocks	Functioning Prior to Refueling, Operations

See Chapters 3.1 → 3.9 (except for 3.6)

See Chapter 3.7

	<u>Test</u>	<u>Frequency</u>
11. Service Water System	Functioning	Each Refueling Shutdown
12. Fire Protection Pump and Power Supply	Functioning	Monthly
13. Spray Additive Tank <i>SE 3.4.4.8</i>	NaOH Concent	<u>Monthly</u> 184 days
14. Accumulator <i>28.11.6</i>	Boron Concentration	Bi-Monthly
15. Primary System Leakage	Evaluate	Daily
16. Diesel Fuel Supply	Fuel Inventory	Daily
17. Spent Fuel Pit	Boron Concentration	Monthly
18. Secondary Coolant Samples	Gross Activity	72 hours (2) (3)
19. Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown

Notes:

- (1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.
- (2) Not required during a cold or refueling shutdown.
- (3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.
- (4) When BAST is required to be operable.

See Chapters 3.4, 3.5, 3.7, 3.8, and 3.3

31.ix

SR 3.6.7.1
SR 3.6.7.2

4.4 Containment Tests

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within specified values.

Specification

4.4.1 Integrated Leakage Rate Test

4.4.1.1 Definitions

31.iii

SR 3.6.1.1

Pa (psig) is the containment vessel design pressure of 60 psig.

Pt (psig) is the containment vessel reduced test pressure for periodic testing.

Lt (weight percent/24 hours) is the maximum allowable leakage rates of the containment vessel test atmosphere at pressure Pt.

La (weight percent/24 hours) is the maximum allowable leakage rate of the containment vessel test atmosphere at pressure Pa; 0.2%/24 hrs.

1.1

Lam and Ltm (weight percent/24 hours) are the total measured containment leakage rates of the containment vessel test atmosphere at pressures Pa and Pt respectively.

4.4.1.2 Pretest Requirements

31.iii

SR 3.6.1.1

- a. A visual examination of the accessible interior and exterior surfaces of the containment structure shall be performed to uncover any evidence of structural deterioration which may affect either the containment structure integrity or leak-tightness. If there is evidence of structural deterioration, integrated leak rate testing shall not be performed until appropriate corrective action has been taken. Except for repairs to correct structural deterioration, however, no repairs or adjustments shall be made during the period between the initiation of the inspection and the performance of the test.
- b. Closure of containment isolation valves shall be accomplished by normal operation and without any preliminary exercising or adjustments.

4.4.1.3

Conduct of Tests

- a. All integrated leak rate tests shall be conducted in accordance with the provisions of American National Standard N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors, March 16, 1972.
- b. The accuracy of each integrated leak rate test shall be verified by a supplemental test which confirms the accuracy of the test instrumentation and calculational methods by determining a leak rate which is within 0.25Lt of the test result. If results are not within 0.25Lt the reason shall be determined, corrective action taken and a successful supplemental test performed.
- c. Integrated leak rate tests shall be conducted at an initial pressure (beginning of test) $P_t \geq 35$ psig.
- d. If during the test, including the supplemental test, potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the test not meeting the acceptance criteria, the test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and an integrated leak rate test performed.

(31.iii)

SR 3.6.1.1

4.4.1.4 Acceptance Criteria

- 31.iii
SR 3.6.1.1
- a. The leakage rate L_{tm} shall be $<0.75 L_t$ at P_t . P_t is defined as the containment vessel reduced test pressure which is greater than or equal to 35 psig. L_{tm} is defined as the total measured containment leakage rate at pressure P_t . L_t is defined as the maximum allowable leakage rate at pressure P_t .
- b. L_t shall be determined as $L_t = L_a \left(\frac{P_t}{P_a} \right)^{1/2}$ which equals .1528 percent weight per day at 35 psig. P_a is defined as the calculated peak containment internal pressure related to design basis accidents which is greater than or equal to 60 psig. L_a is defined as the maximum allowable leakage rate at P_a which equals .2 percent weight per day.
- c. The leakage rate at P_a (L_{am}) shall be $<0.75 L_a$. L_{am} is defined as the total measured containment leakage rate at pressure P_a .

4.4.1.5 Test Frequency

- 31.iii
SR 3.6.1.1
- a. A set of three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted in the final year of the 10-year service period or one year before or after the final year of the 10-year service period provided:
- i. the interval between any two Type A tests does not exceed four years,
 - ii. following each in-service inspection, the containment airlocks, the steam generator inspection/maintenance penetration, and the equipment hatch are leak tested prior to returning the plant to operation, and
 - iii. any repair, replacement, or modification of a containment barrier resulting from the inservice inspections shall be followed by the appropriate leakage test.

b. If any test fails to meet the acceptance criteria of 4.4.1.4.a the test schedule for subsequent regularly scheduled inservice tests shall be submitted to the Commission for review and approval.

c. If two consecutive tests fail to meet the acceptance criteria of 4.4.1.4.a, a retest shall be performed at each refueling shutdown or approximately every 18 months, whichever comes first, until two consecutive tests meet the acceptance criteria of 4.4.1.4.a, after which time the retest schedule of 4.4.1.5.a may be resumed.

31.iii

SR 3.4.1.1

4.4.1.6 Additional Requirements

a. A summary technical report shall be submitted to the Commission after the conduct of each integrated leak rate test. Information on any valve closure malfunction or valve leakage that requires corrective action before the test shall be included in the report.

4.4.2 Local Leak Detection Tests

4.4.2.1 Test

a. Local leakage rate tests shall be performed at intervals specified in 4.4.2.4 below and at a pressure of not less than 60 psig.

31.iii

SR 3.4.1.1

- b. The local leakage rate shall be measured for each of the following components:
 - i. Containment penetrations that employ resilient seals, gaskets, or sealant compounds, piping penetrations with expansion bellows and electrical penetrations with flexible metal seal assemblies.
 - ii. Air lock and equipment door seals.
 - iii. Fuel transfer tube.
 - iv. Isolation valves on the testable fluid systems lines penetrating the containment.
 - v. Other containment components, which require leak repair in order to meet the acceptance criterion for any integrated leakage rate test.

31.iii

SR 3.6.1.1

4.4.2.2 Acceptance Criterion

Containment isolation boundaries are inoperable from a leakage standpoint when the demonstrated leakage of a single boundary or cumulative total leakage of all boundaries is greater than 0.60 La.

4.4.2.3 Corrective Action

- a. If at any time it is determined that the total leakage from all penetrations and isolation boundaries exceeds 0.60 La, repairs shall be initiated immediately.

CCO 3.6.1

31.iv

b. If repairs are not completed and conformance to the acceptance criterion of 4.4.2.2 is not demonstrated within 48 hours, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion.

CCO 3.6.1

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c. If it is determined that the leakage through a mini-purge supply and exhaust line is greater than 0.05 La an engineering evaluation shall be performed and plans for corrective action developed.

SR 3.6.3.4

CCO 3.6.3

31.vi

4.4.2.4 Test Frequency

- a. Except as specified in b. and c. below, individual penetrations and containment isolation valves shall be tested in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.
- b. The containment equipment hatch, fuel transfer tube, steam generator inspection/maintenance penetration, and shutdown purge system flanges shall be tested at each refueling shutdown or after each use, if that be sooner.

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SR 3.3.1.1

SR 3.6.2.1
SR 3.6.2.2
LCO 3.6.2

31.iii

31.v

c. The containment air locks shall be tested at intervals of no more than six months by pressurizing the space between the air lock doors. In addition, following opening of the air lock door during the interval, a test shall be performed by pressurizing between the dual seals of each door opened, within 48 hours of the opening, unless the reactor was in the cold shutdown condition at the time of the opening or has been subsequently brought to the cold shutdown condition. A test shall also be performed by pressurizing between the dual seals of each door within 48 hours of leaving the cold shutdown condition, unless the doors have not been open since the last test performed either by pressurizing the space between the air lock doors or by pressurizing between the dual door seals.

- c. Visual inspection shall be made for excessive leakage from components of the system. Any significant leakage shall be measured by collection and weighing or by an equivalent method.

4.4.3.2 Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.4.3.3 Correction Action

- a. Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.4.3.2.
- b. If repairs are not completed within 24 hours, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion of 4.4.3.2 is satisfied.

4.4.3.4 Test Frequency

Tests of the recirculation heat removal system shall be conducted at intervals not to exceed 12 months.

SR 3.6.1.2 4.4.4 Tendon Stress Surveillance

~~4.4.4.1 Inspection for Broken Wires~~

- ~~a. Fourteen specific tendons, equally spaced around the~~

31.i

See
Chapter
5.0

containment shall be inspected periodically for the presence of broken wires.

- b. The inspection intervals, measured from the date of the initial structural test, shall be as follows:

6 months

1 year

3 years

8 years and 5 years intervals thereafter.

- c. The acceptance criteria for the inspection are that no more than a total of 38 wires (in 14 tendons) are broken and that not more than 6 broken wires exist in any one tendon. If more than 38 broken wires are found, all tendons shall be inspected. If inspection reveals more than 5% of the total wires broken, the reactor shall be shut down and depressurized.

- d. If more than 20 wires (in 14 tendons) have been broken since the last inspection, all tendons shall be inspected. If inspection reveals more than 5% of the total wires broken, the reactor shall be shut down and depressurized.

- e. If as many as 6 broken wires are found in any one tendon, four immediately adjacent tendons (two on each side of

31.1

the tendon containing 6 broken wires) shall be inspected. The accepted criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

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4.4.4.2 Pre-Stress Confirmation Test

- a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.
- b. Before reseating a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each containment isolation valve shall be demonstrated to be OPERABLE in accordance with the Ginna Station Pump and Valve Test program submitted in accordance with 10 CFR 50.55a.

SR 3.4.3.3

31.vii

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

Addressed in chapter 3.3

4.4.6.2 The response time of each containment isolation valve shall be demonstrated to be within its limit at least once per 18 months. The response time includes only the valve travel time for those valves which the safety analysis assumptions take credit for a change in valve position in response to a containment isolation signal.

SR 3.4.3.5

31.viii

Addressed in
chapter 3.3

4.4.7 Containment Hydrogen Monitors

- 4.4.7.1 Demonstrate that two hydrogen monitors are operable at least daily by verifying that the unit is on or in standby.
- 4.4.7.2 At least once per quarter perform a channel calibration using two sample gases containing known concentrations of hydrogen.

Basis:

The containment is designed for an accident pressure of 60 psig. (1) While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure. The maximum temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286° F.

Prior to initial operation, the containment was strength tested at 69 psig and then was leak tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.1% per 24 hours at 60 psig. This leakage rate was believed consistent with the construction of the containment, (2) which is equipped with independent leak-testable penetrations and contains channels over all containment liner welds, which were independently leak tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.20% per 24 hours at 60 psig. With this leakage rate and with minimum containment engineered safeguards operating (i. e., either 2 filter units and no spray, or 1 filter unit and 1 spray, or no filter units and 2 sprays) the public exposure would be well below 10 CFR 100 values in the event of the design basis accident. (3)

Performance of the integrated leakage rate test provides an over-all assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 35 psig for the integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leakage rate test at 35 psig.

The Specification also allows for possible deterioration of the leakage rate between tests, by requiring that the total measured leakage rate be only 75% of the maximum allowable leakage rate.

The duration and methods for the integrated leakage rate test established by ANSI N45.4-1972 provide a minimum level of accuracy and allow for daily cyclic variation in temperature and thermal radiation. The frequency of the integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns. Refueling shutdowns are scheduled at approximately one year intervals.

The specified frequency of integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the use of weld channels to test the leaktightness of the welds during erection, (b) conformance of the complete containment to a 0.1% per day leak rate at 60 psig during preoperational testing, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60 La) of the total leakage that is specified as acceptable. Third is the tendon stress surveillance program, which provides assurance that an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage of 0.60 La from penetrations and isolation boundaries is that only a portion of the allowable integrated leakage rate should be from those sources in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. Because most leakage during an integrated leak rate test occurs through penetrations and isolation valves, and because for most penetrations and isolation valves a smaller leakage rate would result from an integrated leak test than from a local test, adequate assurance of maintaining the integrated leakage rate within the specified limits is provided. The limiting leakage rates from the Recirculation Heat Removal Systems are judgement values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test

pressure, 350 psig, achieved either by normal system operation or by hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return lines and the reactor coolant drain tank piping connections to the residual heat removal system of 100 psig gives an adequate margin over the highest pressure within the lines after a design basis accident. (4)

A recirculation system leakage of 2 gal. /hr will limit offsite exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident. The dose calculated as a result of this leakage is 7.7 mr for a 2-hr exposure at the site boundary. (5)

In case of failure to meet the acceptance criteria for leakage from the residual heat removal system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor. The times allowed for repairs are consistent with the times developed in Specification 3.3.

The tendon surveillance program is based on assuring that, on the average, the load-carrying capability of the tendons is maintained at approximately 95% design.

See Chapter 5.0

See Chapter 5.0

The pre-stress confirmation test provides a direct measure of the load-carrying capability of the tendon.

If the surveillance program indicates by extensive wire breakage or tendon stress relation that the pre-stressing tendons are not behaving as expected, the situation will be evaluated immediately. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. The containment is provided with two readily removable tendons that might be useful to such a study. In addition, there are 40 tendons, each containing a removable wire which will be used to monitor for possible corrosion effects.

Operability of the containment isolation boundaries ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Performance of cycling tests and verification of isolation times associated with automatic containment isolation valves is covered by the Pump and Valve Test Program. Compliance with Appendix J to 10 CFR 50 is addressed under local leak testing requirements.

References:

- (1) UFSAR Section 3.1.2.2.7
- (2) UFSAR Section 6.2.6.1
- (3) UFSAR Section 15.6.4.3
- (4) UFSAR Section 6.3.3.8
- (5) UFSAR Table 15.6-9
- (6) FSAR Page 5.1.2-28
- (7) North-American-Rockwell Report 550-x-32, Autonetics Reliability Handbook, February 1963.
- (8) FSAR Page 5.1.2-28

Addressed
in Chapter
3.5

b. The system test will be considered satisfactory if control board indication and visual observations indicate that all valves have received the Safety Injection signal and have completed their travel. The proper sequence and timing of the rotating components are to be verified in conjunction with Section 4.6.1 b.

4.5.1.2 Containment Spray System

SR 3.4.6.13

a. System tests shall be performed at each reactor refueling interval. The test shall be performed with the isolation valves, in the spray supply lines, at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

SR 3.4.6.2
SR 3.6.6.1
32.vii

b. The spray nozzles shall be checked for proper functioning at least every ~~five~~ ^{ten} years.

SR 3.6.6.18
32.viii

c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.5.2 Component Tests

4.5.2.1 Pumps

SR 3.4.6.7

32.ii

a. Except during cold or refueling, shutdowns the safety injection pumps, residual heat removal pumps, and containment spray pumps shall be started at intervals not to exceed one month. The pumps shall be tested prior to startup if the time since the last test exceeds 1 month.

SR 3.6.6.7

32.11

See also chapter 3.5

b. Acceptable levels of performance for the pumps shall be that the pumps start, operate, and develop the minimum discharge pressure for the flows listed in the table below:

PUMPS	RECYCLE FLOW RATE	DISCHARGE PRESSURE
Containment Spray Pumps	35 gpm	240 psig
Residual Heat Removal Pumps	[200 gpm] 450 gpm	[140 psig] 138 psig
Safety Injection Pumps	[50 gpm] 150 gpm	[1420 psig] 1356 psig

Notes

(1)

(2)

Table 4.5-1

Notes

- (1) Items in square brackets are effective until the installation of the new residual heat removal minimum flow recirculation system.
- (2) Items in square brackets are effective until installation of the new safety injection minimum flow recirculation system.

4.5.2.2 Valves

SR 3.6.6.12
SR 3.6.6.16
SR 3.6.6.17

32.x

a. Except during cold or refueling shutdowns the spray additive valves shall be tested at intervals not to exceed ^{24 months} one month. With the pumps shut down and the valves upstream and downstream

32.x

of the spray additive valves closed, each valve will be opened and closed by operator action. This test shall be performed prior to startup if the time since the last test exceeds one month.

See Chapter 3.5

3. The accumulator check valves shall be checked for operability during each refueling shutdown.

4.5.2.3 Air Filtration System

SR 3.6.6.4
SR 3.6.6.14

4.5.2.3.1 At least once every 18 months or after every 720 hours of charcoal filtration system operation since the last test, or following painting, fire or chemical release in any ventilation zone communicating with the system, the post accident charcoal system shall have the following conditions demonstrated.

SR 3.6.6.10

See also Chapter 5.0

- a. The pressure drop across the charcoal adsorber bank is less than 3 inches of water at design flow rate ($\pm 10\%$).
- b. In place Freon testing, under ambient conditions, shall show at least 99% removal.
- c. The iodine removal efficiency of at least one charcoal filter cell shall be measured. The filter cell to be tested shall be selected randomly from those cells with the longest in-bank residence time. The minimum acceptable value for filter efficiency is 90% for removal of methyl iodide when tested at at least 286°F and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

4.5.2.3.2 After each replacement of a charcoal drawer or after any structural maintenance on the housing for the post accident charcoal system, the condition of Specification 4.5.2.3.1.b shall be demonstrated for the affected portion of the system.

SR 3.6.6.5
See also Chapter 5.0

4.5.2.3.3 At least every 18 months or following painting, fire, or chemical release in any ventilation zone communicating with the system, the containment recirculation system shall have the following conditions demonstrated.

SR 3.6.6.6

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- a. The pressure drop across the HEPA filter bank is less than 3 inches of water at design flow rate ($\pm 10\%$).
- b. In place thermally generated DOP testing of the HEPA filters shall show at least 99% removal.

See also Chapter 5.0

4.5.2.3.4 After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on a housing for the containment recirculation system, the condition of Specification 4.5.2.3.3.b shall be demonstrated for the affected portion of the system.

4.5.2.3.5 Except during cold or refueling shutdowns the post accident charcoal filter isolation valves shall be tested at intervals not greater than one month to verify operability and proper orientation and flow shall be maintained through the system for at least 15 minutes. The test shall be performed prior to startup if the time since the last test exceeds 1 month.

SR 3.6.6.3

SR 3.6.6.12

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See also Chapter 5.0

4.5.2.3.9 Except during cold or refueling shutdowns the automatic initiation of the control room emergency air treatment system shall be tested at intervals not to exceed one month to verify operability and proper orientation and flow shall be maintained through the system for at least 15 minutes. The test shall be performed prior to startup if the time since the last test exceeds one month.

Addressed in
chapter
2.2

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a Safety Injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The annual systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action

and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump, circuit breakers, and automatic circuitry.⁽¹⁾

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order and develop the minimum required pressure to meet accident conditions.⁽²⁾ The minimum discharge pressure values listed in Table 4.5-1 are based on an assumed degradation of the pump head-capacity (characteristic) curve adjusted to water temperature of 60°F as follows:

Containment Spray Pumps	5%*
Residual Heat Removal Pumps	5%*
Safety Injection Pumps	3%*

*Percentage is based on the head at the best efficiency point of flow.

The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required) and would result in increased wear over long periods of time.

Other systems that are also important to the emergency cooling function are the accumulators, the component cooling system, the service water system and the containment fan coolers. The accumulators are a passive safeguard. In accordance with the specifications, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance. The reactor coolant drain tank pumps operate intermittently during reactor operation, and thus are also monitored for satisfactory performance.

The air filtration portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. The pressure drop, filter efficiency, and valve operation test frequencies will assure that the system can operate to meet its design function under accident conditions. As the adsorbing charcoal is normally isolated, the test schedule, related to hours of operation as well as elapsed time, will assure that it does not degrade below the required adsorption.

~~efficiency. The test conditions for charcoal sample adsorbing efficiency are those which might be encountered under an accident situation.⁽³⁾~~

See Chapter 3.2

The control room air treatment system is designed to filter the control room atmosphere (recirculation and intake air) during control room isolation conditions. HEPA filters are installed before the charcoal filters to remove particulate matter and prevent clogging of the iodine adsorbers. The charcoal filters reduce the airborne radioiodine in the control room. Bypass leakage must be at a minimum in order for these filters to perform their designed function. If the performances are as specified the calculated doses will be less than those analyzed.⁽⁴⁾

~~Retesting of the post accident charcoal system or the control room emergency air treatment system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.~~

~~Testing of the air filtration systems will be, to the extent it can, given the configuration of the systems, in accordance with ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems."~~

References:

- (1) UFSAR Section 6.3.5.2
- (2) UFSAR Figures 15.6-12 and 15.6-13
- (3) UFSAR Section 6.5.1.2.4
- (4) UFSAR Section 6.4.3.1

3.1.4 Maximum Coolant Activity Specifications

3.1.4.1 Whenever the reactor is critical or the reactor coolant average temperature is greater than 500°F:

- See Chapter 3.4
- a. The total specific activity of the reactor coolant shall not exceed $84/\bar{E}$ $\mu\text{Ci/gm}$, where \bar{E} is the average beta and gamma energies per disintegration in Mev.
 - b. The I-131 equivalent of the iodine activity in the reactor coolant shall not exceed 0.2 $\mu\text{Ci/gm}$.
 - c. The I-131 equivalent of the iodine activity on the secondary side of a steam generator shall not exceed 0.1 $\mu\text{Ci/gm}$.
- See Chapter 3.4

See Chapter 3.4

3.1.4.2 If the limit of 3.1.4.1.a is exceeded, then be subcritical with reactor coolant average temperature less than 500°F within 8 hours.

See Chapter 3.4

3.1.4.3 a. If the I-131 equivalent activity in the reactor coolant exceeds the limit of 3.1.4.1.b but is less than the allowable limit shown on Figure 3.1.4-1, operation may continue for up to 168 hours.

The reactor may be taken critical or reactor coolant average temperature may be increased above a 500°F with the I-131 equivalent activity greater than the limit of 3.1.4.1.b as long as the provisions of this paragraph are met.

- b. If the I-131 equivalent activity exceeds the limit of 3.1.4.1.b for more than 168 hours during one continuous time interval or exceeds the limit shown on Figure 3.1.4-1, be subcritical with reactor coolant average temperature less than 500°F within 8 hours.
- c. If the I-131 equivalent activity exceeds the limit of 3.1.4.1.b, then perform sampling and analysis as required by Table 4.1-4, item 4a, until the activity is reduced to less than the limit of 3.1.4.1.b.

3.1.4.4 If the limit of 3.1.4.1.c is exceeded, then be at hot shutdown within ⁶ hours and in cold shutdown within the following ³⁰ ~~32~~ hours.

Basis:

The total activity limit for the primary system corresponds to operation with the plant design basis of 1% fuel defects. (1)

Radiation shielding and the radioactive waste disposal systems

See
Chapter 2.4

were designed for operation with 1% defects⁽²⁾. The limit for secondary iodine activity is conservatively established with respect to the limits on primary system iodine activity and primary-to-secondary leakage (Specification 3.1.5.2). If the activity should exceed the specified limits following a power transient the major concern would be whether additional fuel defects had developed bringing the total to above 1% defects. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the volume control tank gases to the gas decay tanks.

See
Chapter
3.4

The specified activity limits provide protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the analysis of a steam generator tube rupture accident.⁽³⁾

The 500°F temperature in the specification corresponds at saturation to 681 psia, which is below the set point of the secondary side relief valves. Therefore, potential primary to secondary leakage at a temperature below 500°F can be contained by closing the steam line isolation valves.

References:

- (1) FSAR Table 9.2-5
- (2) FSAR Section 11.1.3
- (3) Letter dated September 24, 1981 from Dennis M. Crutchfield, USNRC, to John E. Maier, RG&E.

See Chapter
3.6

- b. One containment spray pump may be inoperable provided the pump is restored to operable status within 3 days.
- c. Any valve or piping in a system, required to function during accident conditions, may be inoperable provided it is restored to operable status within 72 hours.
- d. One post accident charcoal filter unit and/or its associated fan cooler may be inoperable provided the unit is restored to operable status within 7 days.
- e. The spray additive system may be inoperable for a period of no more than 3 days provided that both charcoal filter units are operable.

3.3.3 Component Cooling System

3.3.3.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

CCO 3.7.7

- a. Both component cooling pumps are operable.
- b. Both component cooling heat exchangers are operable.
- c. All valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable. (i.e., loop hoister)

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3.3.3.2 The requirements of 3.3.3.1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.3.1 within the time period

CCO 3.7.7

13. xvii

specified, the reactor shall be in hot shutdown within the next 6 hours. ^{and} If the requirements of 3.3.3.1 are not satisfied within an additional 48 hours, the reactor shall be in the cold shutdown condition within the following 30 hours. If neither component cooling water loop is operable, ^{or the loop header is inoperable} the reactor shall be maintained below a reactor coolant system temperature of 350°F instead of at cold shutdown and corrective action shall be initiated to restore a component cooling water loop to operable status as soon as possible.

LCO 3.7.7

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a. One component cooling pump may be out of service provided the pump is restored to operable status within ⁷²~~24~~ hours.

b. One heat exchanger or other passive component may be out of service provided the system may still operate at 100% capacity and repairs are completed within ~~24 hours~~ ^{31 days}

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3.3.4 Service Water System

3.3.4.1 The reactor shall not be taken above cold shutdown unless the following conditions are met:

LCO 3.7.8

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a. At least two service water pumps, one on bus 17 and one on bus 18, ^{Six sets of isolation valves} and one loop header are operable.

b. All valves, interlocks, and piping associated with the operation of two pumps are operable.

3.3.4.2 Any time that the conditions of 3.3.4.1 above cannot be met, the reactor shall be placed in hot shutdown within ^{components shall be restored within 72 hours on the} 6 hours and in cold shutdown within an additional 30 hours.

LCO 3.7.8

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3.3.5 Control Room Emergency Air Treatment System

3.3.5.1 The ~~RCS temperature shall not be at or above 350°F~~
~~unless the~~ control room emergency air treatment system ^{Shut off}

LCO 3.7.9

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~~is~~ operable ^{in MODES 1 → 6 and during movement of irradiated fuel assemblies}

3.3.5.2 The requirements of 3.3.5.1 may be modified to allow ^{filter drain} the control room emergency air treatment system to be

LCO 3.7.9

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inoperable for a period of 48 hours. If the system is not made operable within those 48 hours, the reactor shall be placed in hot shutdown within the next 6 hours and ~~the RCS temperature less than 350°F~~ ³⁰ in an additional ¹² hours. ← ^{Cold shutdown}

Basis

The normal procedure for starting the reactor is, first to heat the

If one isolation damper is inoperable, replace within 7 days.

If both isolation dampers are inoperable, enter LCO 3.0.3 and immediately stop all fuel movement.

See Chapter 3.6

dundancy for certain ranges of break sizes. (2)

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus post accident charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, only two of the four fan-coolers are required to remove heat lost from equipment and piping within containment. (3) In the event of a Design Basis Accident, any one of the following will serve to reduce airborne iodine activity and maintain doses within the values calculated in the FSAR: (1) two containment spray pumps and sodium hydroxide addition, (2) two fan-coolers and two post accident charcoal filters, or (3) one containment spray pump and sodium hydroxide addition plus one fan-cooler and one post accident charcoal filter. (4) In addition, the containment integrity analysis assumes that one containment spray pump and two fan-coolers operate to reduce containment pressure following a Design Basis Accident. (9) Because of the difficulty of access to make repairs to a fan-cooler and because of the low probability of a Design Basis Accident during that time, an additional seven days operation with an inoperable fan-cooler is permitted. The containment spray pumps and spray additive system are located outside containment and are, therefore, less difficult to repair. Therefore, three days with an inoperable containment spray pump or spray additive system is deemed acceptable.

~~The Component Cooling System is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. (5) In addition, if during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected. (6)(7)~~

The facility has four service water pumps. Only one is needed during the injection phase, and two are required during the recirculation phase of a postulated loss-of-coolant accident.⁽⁸⁾ The control room emergency air treatment system is designed to filter the control room atmosphere during periods when the control room is isolated and to maintain radiation levels in the control room at acceptable levels following the Design Basis Accident.⁽⁹⁾ Reactor operation may continue for a limited time while repairs are being made to the air treatment system since it is unlikely that the system would be needed. Technical Specification 3.3.5 applies only to the equipment necessary to filter the control room atmosphere. Equipment necessary to initiate isolation of the control room is covered by another specification.

The limits for the accumulator pressure and volume assure the required amount of water injection during an accident, and are based on values used for the accident analyses. The indicated level of 50% corresponds to 1108 cubic feet of water in the accumulator and the indicated level of 82% corresponds to 1134 cubic feet.

The limitation of no more than one safety injection pump to be operable when overpressure protection is being provided by a RCS vent of ≥ 1.1 sq. in. insures

See
Chapter
3.5

References

- (1) Deleted
- (2) UFSAR Section 6.3.3.1
- (3) UFSAR Section 6.2.2.1
- (4) UFSAR Section 15.6.4.3
- (5) UFSAR Section 9.2.2.4
- (6) UFSAR Section 9.2.2.4
- (7) Deleted
- (8) UFSAR Section 9.2.1.2
- (9) UFSAR Section 6.2.1.1 (Containment Integrity) and UFSAR Section 6.4 (CR Emergency Air Treatment)
- (10) Westinghouse Report, "R.E. Ginna Boric Acid Storage Tank Boron Concentration Reduction Study" dated Nov. 1992 by C.J. McHugh and J.J. Spryshak

3.4 TURBINE CYCLEApplicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity, and to define the Auxiliary Feedwater System and supporting Service Water System operation as necessary to ensure the capability to remove core decay heat. The Standby Auxiliary Feedwater System provides additional assurance of capability to remove core decay heat should the Auxiliary Feedwater System be unavailable.

3.4.1 MAIN STEAM SAFETY VALVES

Specification

Except during testing of the main steam safety valves, with the RCS temperature at or above 350°F, a minimum turbine cycle code approved steam relieving capability of eight (8) main steam safety valves shall be available.

In MODE 3

14.i

LCO 3.7.1

Action

With one or more main steam code safety valves inoperable, restore the inoperable valve(s) to operable status within 4 hours or be in hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

3.4.2 AUXILIARY FEEDWATER

3.4.2.1. MOTOR-DRIVEN AUXILIARY FEEDWATER SYSTEM

Specification

With the RCS temperature at or above 350°F, both motor-driven auxiliary feedwater pumps must be operable, each with an operable flow path from the condensate storage tanks to its respective steam generator.

LCO 3.7.5

Action

- a. With one motor-driven auxiliary feedwater pump inoperable and at least one turbine-driven auxiliary feedwater pump flowpath operable, restore the pump to operable status within 7 days or be in at least hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

to the opposite SG

- b. With both motor-driven ^(both) auxiliary feedwater pumps inoperable, and ~~at least one~~ turbine-driven auxiliary feedwater pump flowpath operable (see 3.4.2.2), or with a motor-driven and turbine-driven pump ~~(or both flow paths)~~ inoperable, restore a pump to operable status within ~~6~~ ⁽¹²⁾ hours or be in at least hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.
If all AFW to one or more SGs is inoperable, restore within 4 hours.
- c. With all auxiliary feedwater pumps inoperable (motor-driven, turbine-driven, and standby), immediately initiate corrective action to restore any of these pumps to operable status as soon as possible.

(14.i)

3.4.2.2 TURBINE-DRIVEN AUXILIARY FEEDWATER SYSTEM

Specification

With the RCS temperature at or above 350°F, the turbine-driven auxiliary feedwater pump associated flow paths from the condensate storage tanks to the steam generators, and flow paths of steam from each steam generator to the pump turbine, must be operable. The turbine-driven auxiliary feedwater pump must be shown to be operable prior to exceeding 5% power.

LCO 3.7.5
SR 3.7.5.2

Action

- a. With the turbine-driven auxiliary feedwater pump and/or both associated flow paths inoperable, restore the pump (and at least one flow path) to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the next 6 hours.
- b. With one associated flow path of the turbine-driven auxiliary feedwater pump inoperable, restore to operable status within 7 days or be in at least hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the next 6 hours.

3.4.2.3 STANDBY AUXILIARY FEEDWATER SYSTEM

Specification

With the RCS temperature at or above 350°F, two standby auxiliary feedwater pumps each with an associated flow path from the service water system to ~~its respective~~ steam generator, shall be operable. ^(both)

LCO 3.7.5
(14.iii)

Action

- a. With one standby auxiliary feedwater pump inoperable restore the pump to operable status within 14 days or be in hot shutdown within the next 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

- b. With both standby auxiliary feedwater pumps inoperable restore at least one standby auxiliary feedwater pump to operable status within 7 days or be in at least hot shutdown within 6 hours and at an RCS temperature less than 350°F in the following 6 hours.

3.4.3 SOURCES OF AUXILIARY FEEDWATER

Specification

- a. With the RCS temperature at or above 350°F, the following sources of auxiliary feedwater shall be operable:

LCO 3.7.4 (1) One or more condensate storage tanks with a minimum of 22,500 gallons of water, and

LCO 3.7.5 (14.1v) (2) Service water as the primary supply to the standby auxiliary feedwater pumps.

Action

- a. With the condensate storage tanks inoperable, within 4 hours either:

1) restore the condensate storage tanks to operable status, or be in at least hot shutdown within the following 6 hours and at an RCS temperature less than 350°F within the following 6 hours, OR.

LCO 3.7.4

(14.v)

2) demonstrate the operability of the service water system as a water supply to the motor-driven and turbine-driven auxiliary feedwater pumps and restore the condensate storage tanks to operable status within 7 days, or be in at least hot shutdown within the following 6 hours and at an RCS temperature less than 350°F within the following 6 hours.

- b. With the service water system to one or both standby auxiliary feedwater pump(s) inoperable, declare the standby auxiliary feedwater pump(s) inoperable and comply with Specification 3.4.2.3.

LCO 3.7.5

(14.1v)

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

In the event of a reactor and turbine trip, together with a loss of offsite power, immediate decay heat removal is effected via the main steam safety valves. The eight main steam safety valves have a total combined rated capability of 6,580,000 lbs/hr. This capability exceeds the total full-power steam flow of 6,577,279 lbs/hr.

Following reactor/turbine trip, the motor-driven auxiliary feedwater system is automatically initiated on low-low level in one steam generator, a Safety Injection signal, or a trip of both main feedwater pumps. The turbine-driven auxiliary feedwater pump is initiated on low-low steam generator level in both steam generators, or a loss of power to electrical buses 11A and 11B. The motor-driven auxiliary feedwater system has two 100% capacity pumps, each normally serving one steam generator.

Their sources of water include the normally-aligned but non-safety-related and non Seismic Category I condensate storage tanks, and the safety-related service water system. The turbine-driven auxiliary feedwater system consists of one 200% capacity pump, two steam supply flow paths (one from each steam generator), a normal source of water from the non-safety-related condensate storage tanks, and a backup source of water from the safety-related service water system.⁽¹⁾

The Ginna Station accident analyses⁽²⁾ assume 200 gpm is delivered to an operable steam generator, in order to remove the required decay heat. The combination of motor-driven and turbine-driven auxiliary feedwater pumps assures operability of the system to meet these requirements, even assuming a single failure.

In the event of a high energy line break outside containment,⁽³⁾ the operability of the motor-driven and turbine-driven auxiliary feedwater systems cannot be ensured, since the systems are not qualified for the ensuing harsh environment. The standby auxiliary feedwater system, which consists of two redundant pumps, a discharge flow path to each steam generator and suction from both loops of the safety-related service water system, performs this function. Operator action from the control room is required to effect operation of the SAFW system. The worst-case analysis, a feedwater line break,⁽⁴⁾ has been performed, and the consequences were found to be acceptable.

The minimum amount of water in the condensate storage tanks is the amount needed to remove decay heat for 2 hours after reactor trip from full power.⁽⁵⁾ An unlimited source for auxiliary feedwater is available using the safety-related service water system.

References:

- (1) UFSAR Section 10.5
- (2) UFSAR Sections 15.2, 15.3, 15.6
- (3) "Effects of High Energy Pipe Breaks Outside the Containment Building" submitted by letter dated November 1, 1973 from K.W. Amish, Rochester Gas and Electric Corporation to A. Giambusso, Deputy Director for Reactor Projects.
- (4) L.D. White, Jr. letter to Mr. D.L. Ziemann, USNRC dated March 28, 1980.
- (5) SEP Topic XV-6, Feedwater System Pipe Breaks, NRC SER dated 9/4/81

3.11 Fuel Handling in the Auxiliary Building

Applicability

Applies to limitations while moving irradiated fuel in the operating floor area of the auxiliary building.

Objective

To limit doses in the event an irradiated fuel assembly is damaged significantly.

Specification

3.11.1 During handling of fuel assemblies in the auxiliary building when either the fuel being handled or the fuel stored in the spent fuel storage pool has decayed less than 60 days since irradiation, the following conditions shall be satisfied:

LCO 2.7.10

21.1

- a. One auxiliary building main exhaust fan shall be operating.
- b. The auxiliary building exhaust fan 1C, which takes suction from the spent fuel storage pool area, shall be operating.
- c. ~~All doors, windows, and other direct openings between the operating floor area and the outside shall be closed except that the personnel door may be opened for access as required.~~
- d. Roughing filters shall be installed at the inlet to the adsorbers.

Shall have a negative pressure with respect to

3.11.1

Amendment No. 19

LCO 3.7.10

21.i

e. Charcoal adsorbers shall be installed in the ventilation system exhaust from the spent fuel storage pool area and shall be operable.

21.ii

3.11.2 Radiation levels in the spent fuel storage area shall be monitored continuously.

21.iii

3.11.3 A load in excess of one fuel assembly and its handling tool shall never be stationed or permitted to pass over storage racks containing spent fuel.

21.iv

3.11.4 The spent fuel pool temperature shall be limited to 150°F.

21.v

3.11.5 The restriction of 3.11.3 above shall not apply to the movement of canisters containing consolidated fuel rods if the spent fuel rack beneath the transported canister contain only spent fuel that has decayed at least 60 days since reactor shutdown.

Basis:

Charcoal adsorbers will reduce significantly the consequences of a refueling accident which considers the clad failure of a single irradiated fuel assembly. Therefore, charcoal adsorbers should be employed whenever recently irradiated fuel is being handled. This requires that the ventilation system should be operating and drawing air through the adsorbers. The only exception to the requirement occurs when the fuel being handled, or any fuel in the storage pool, has decayed at least 60 days since irradiation. The consequences of a fuel handling accident in this case without operation of the charcoal adsorbers is significantly less than the guidelines of 10CFR100.³

Amendment No. 6, 10, 12, 19

Amendment No.

The desired air flow path, when handling irradiated fuel, is from the outside of the building into the operating floor area, toward the spent fuel storage pool, into the area exhaust ducts, through the adsorbers, and out through the ventilation system exhaust to the facility vent. Operation of a main auxiliary building exhaust fan assures that air discharged into the main ventilation system exhaust duct will go through a HEPA and be discharged to the facility vent. Operation of the exhaust fan for the spent fuel storage pool area causes air movement on the operating floor to be toward the pool. Proper operation of the fans and setting of dampers would result in a negative pressure on the operating floor which will cause air leakage to be into the building. Thus, the overall air flow is from the location of low activity (outside the building) to the area of highest activity (spent fuel storage pit). The exhaust air flow would be through a roughing filter and charcoal before being discharged from the facility. The roughing filter protects the adsorber from becoming fouled with dirt; the adsorber removes iodine, the isotope of highest radiological significance, resulting from a fuel handling accident. The effectiveness of charcoal for removing iodine is assured by having a high throughput and a high removal efficiency. The throughput is attained by operation of the exhaust fans. The high removal efficiency is attained by minimizing the amount of iodine that bypasses the charcoal and having charcoal with a high potential for removing the iodine that does pass through the charcoal.



The spent fuel pool temperature is limited to 150°F because if the spent fuel pool cooling system is not at that temperature, sufficient time (approximately 7 hours) is available to provide backup cooling, assuming the maximum anticipated heat load (full core discharge & previously stored fuel), until a temperature of 180°F is reached, the temperature at which the structural integrity of the pool was analyzed and found acceptable.

The requirement of 3.11.5 insures that should a handling accident occur during the movement of a consolidated fuel cannister (as described in 5.4.) the dose at the exclusion area boundary would satisfy the requirements of 10CFR100.

References

- (1) FSAR - Section 9.3-1.
- (2) ANS-5.1 (N 18.6), October 1973
- (3) Letter, J.A. Zwolinski, (USNRC) to R.W. Kober, (RG&E),
November 14, 1984.

- SR 3.7.11.1
- SR 3.7.13.1
- SR 3.7.13.2
- SR 3.7.6.1
- SR 3.7.7.1
- SR 3.7.7.2
- SR 3.7.8.1
- SR 3.7.14.1
- SR 3.7.8.2
- SR 3.7.8.3

28.iii.i

TABLE 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

See chapters 3.1, 3.4, 3.5

	<u>Test</u>	<u>Frequency</u>
1.	Reactor Coolant Chemistry Samples Chloride and Fluoride Oxygen	3 times/week and at least every third day 5 times/week and at least every second day except when below 250°F
2.	Reactor Coolant Boron Boron Concentration	Weekly
3.	Refueling Water Storage Tank Water Sample Boron Concentration	Weekly
4.	Boric Acid Storage Tank Boron Concentration	Twice/Week ⁽⁴⁾
5.	Control Rods Rod drop times of all full length rods	After vessel head removal and at least once per 18 months (1)
6a.	Full Length Control Rod Move any rod not fully inserted a sufficient number of steps in any one direction to cause a change of position as indicated by the rod position indication system	Monthly
6b.	Full Length Control Rod Move each rod through its full length to verify that the rod position indication system transitions occur	Each Refueling Shutdown
7.	Pressurizer Safety Valves Set point	Each Refueling Shutdown
SR 3.7.11.1	8. Main Steam Safety Valves Set point ^{+1%} -3%	Each Refueling Shutdown
	9. Containment Isolation Trip Functioning	Each Refueling Shutdown
28.ii.d	10. Refueling System Interlocks Functioning	Prior to Refueling Operations

See chapter 3.3

SR 3.7.8.4
SR 3.7.8.5

Test

Frequency

	<u>Test</u>	<u>Frequency</u>
11. Service Water System	Functioning	Each Refueling Shutdown
12. Fire Protection Pump and Power Supply	Functioning	Monthly
13. Spray Additive Tank	NaOH Concent	Monthly
14. Accumulator	Boron Concentration	Bi-Monthly
15. Primary System Leakage	Evaluate	Daily
16. Diesel Fuel Supply	Fuel Inventory	Daily
17. Spent Fuel Pit	Boron Concentration	Monthly
18. Secondary Coolant Samples	Gross Activity	72 hours (2) (3) 31 days
19. Circulating Water Flood Protection Equipment	Calibrate	Each Refueling Shutdown

28.ii.M

28.ii.a
SR 3.7.12.1

SR 3.7.14.1
28.ii.M

See Chapters 3.5, 2.6, 2.8

See Chapter 3.3

Notes:

(1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

(2) Not required during a cold or refueling shutdown.

28.ii.n

(3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.

(4) When BAST is required to be operable.

See Chapter 3.1

4.5.2.3.6 At least once every 18 months or after every 720 hours of charcoal filtration system operation since the last test, or following painting, fire or chemical release in any ventilation zone communicating with the system, the control room emergency air treatment system shall have the following conditions demonstrated.

- a. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6" of water at design flow rate (+ 10%).
- b. In place Freon testing, under ambient conditions, shall show at least 99% removal.
- c. In place thermally generated DOP testing of the HEPA filters shall show at least 99% removal.
- d. The results of laboratory analysis on a carbon sample shall show 90% or greater radioactive methyl iodide removal when tested at at least 125°F and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

4.5.2.3.7 After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the HEPA housing for the control room emergency air treatment system, the condition of Specification 4.5.2.3.6.c shall be demonstrated for the affected portion of the system.

4.5.2.3.8 After each replacement of a charcoal drawer or after any structural maintenance on the charcoal housing for the control room emergency air treatment system, the condition of Specification 4.5.2.3.6.b shall be demonstrated for the affected portion of the system.

SR 3.7.9.2
SR 3.7.9.1

See also
C 5.0

4.5.2.3.9 Except during cold or refueling shutdowns the automatic initiation of the control room emergency air treatment system shall be tested at intervals not to exceed one month to verify operability and proper orientation and flow shall be maintained through the system for at least 15 minutes. The test shall be performed prior to startup if the time since the last test exceeds one month.

SR 3.7.9.3
 SR 3.7.9.2
 SR 3.7.9.1

32.xii

See Chapters 3.5 and 3.6

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a Safety Injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The annual systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting a test signal is applied to initiate automatic action

See Chapter 3.4

efficiency. The test conditions for charcoal sample adsorbing efficiency are those which might be encountered under an accident situation.⁽³⁾

The control room air treatment system is designed to filter the control room atmosphere (recirculation and intake air) during control room isolation conditions. HEPA filters are installed before the charcoal filters to remove particulate matter and prevent clogging of the iodine adsorbers. The charcoal filters reduce the airborne radiiodine in the control room. Bypass leakage must be at a minimum in order for these filters to perform their designed function. If the performances are as specified the calculated doses will be less than those analyzed.⁽⁴⁾

Retesting of the post accident charcoal system or the control room emergency air treatment system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.

Testing of the air filtration systems will be, to the extent it can, given the configuration of the systems, in accordance with ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems."

References:

- (1) UFSAR Section 6.3.5.2
- (2) UFSAR Figures 15.6-12 and 15.6-13
- (3) UFSAR Section 6.5.1.2.4
- (4) UFSAR Section 6.4.3.1

4.7 Main Steam Isolation Valves

Applicability

Applies to periodic testing of the main steam isolation valves.

Objective

To verify the ability of the main steam isolation valves to close upon signal.

MODES 1, 2, and 3

Specification

SR 3.7.2.2
SR 3.7.2.1

34.i

The main steam isolation valves shall be tested at each refueling interval. Closure time of five seconds or less shall be verified. The valves are tested under no flow and at no

load conditions. ← The MSIVs shall be tested on an actual or simulated close signal once every 24 months

Basis

The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.

References:

- FSAR - Section 10.4
- FSAR - Section 14.2.5

35.vi - SR 3.7.5.1

4.8 AUXILIARY FEEDWATER SYSTEMS

Applicability

Applies to periodic testing requirements of the turbine-driven, motor-driven auxiliary feedwater pumps, and of the standby auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and the standby auxiliary feedwater system and their ability to respond properly when required.

Specification

4.8.1 Except when below 350°F each motor-driven auxiliary feedwater pump, unless it is declared inoperable without testing, will be started at intervals not to exceed ~~one month~~ ^{three} and a flowrate of ~~200 gpm~~ established.

4.8.2 Except when below 350°F the steam turbine-driven auxiliary feedwater pump, unless it is declared inoperable without testing, will be started at intervals not to exceed ~~one month~~ ^{three} and a flowrate of ~~400 gpm~~ established. If one discharge flow path is inoperable in accordance with Specification 3.4.2.2, a flow of 200 gpm must be established. Once the inoperable discharge flow path is returned to operable status, a flow of 400 gpm must be established within 72 hours thereafter.

4.8.3 Except when below 350°F the auxiliary feedwater pumps suction, discharge, ~~and crossover motor operated valves~~ shall be exercised at intervals ~~not to exceed one month~~.

4.8.4 Except when below 350°F each standby auxiliary feedwater pump, unless it is declared inoperable without testing, will be started at intervals not to exceed ~~one month~~ ^{Specified in the Inservice Testing Program} and a flowrate of ~~200 gpm~~ established.

4.8.5 Except when below 350°F the ~~suction, discharge, and crossover motor operated valves~~ for the standby auxiliary feedwater pumps shall be exercised at intervals ~~not to exceed one month~~.

4.8.6 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. These tests shall be performed prior to exceeding 5% power during startup ~~if the time since the last test exceeds one month~~.

4.8.7 At least once per ~~18~~ ²⁴ months, control of the standby auxiliary feed system pumps and valves from the control room will be demonstrated.

Specified in the IST Program

4.8.8 At least once per ⁽²⁴⁾ months during shutdown:

See 3.7.5.5

a. Verify that each automatic valve in the flow path for each auxiliary feedwater pump actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

See 3.7.5.6

b. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

See Chapter 3.2

4.8.9 Each instrumentation channel shall be demonstrated operable by the performance of the Channel Check, Channel Calibration, and Channel Functional Test operations for the modes and at the frequencies shown in Table 4.1-1.

4.8.10 The response time of each pump and valve required for the operation of each "train" of auxiliary feedwater shall be demonstrated to be within the limit of 10 minutes at least once per 18 months.

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet minimum required flowrates. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.⁽¹⁾ Proper functioning of the steam turbine admission valve and the feedwater pumps start will demonstrate the integrity of the steam driven pump.⁽²⁾

Monthly testing of the standby auxiliary feedwater pumps by supplying water from a condensate supply tank to the steam generators will verify their ability to meet minimum required flowrates.

The standby auxiliary feedwater pumps would be used only if all three auxiliary feedwater pumps were unavailable.⁽³⁾ One of the two standby pumps would be sufficient to meet decay heat removal requirements. Proper functioning of the suction valves from the service water system, the discharge valves, and the crossover valves will demonstrate their operability. The operability of the standby auxiliary feedwater pump flow paths between the pumps and the steam generators is demonstrated using water from the test tank. Testing of the auxiliary feedwater pumps using their primary source of water supply will verify the operability of the auxiliary feedwater flow path.

Verification of correct operation will be made both from instrumentation within the main control room and by direct visual observation of the pumps.

References:

- (1) FSAR - Section 10.5
- (2) FSAR - Sections 15.2, 15.3, 15.6
- (3) "Effects of High Energy Pipe Breaks Outside the Containment Building" submitted by letter dated November 1, 1973 from K.W. Amish, Rochester Gas and Electric Corporation to A. Giambusso, Deputy Director for Reactor Projects, U.S. Atomic Energy Commission.

4.11 Refueling

Applicability

Applies to refueling and to fuel handling in the spent fuel pool.

Specification

4.11.1 Spent Fuel Pit Charcoal Adsorber System

4.11.1.1 Within 60 days prior to any operation of the spent fuel pool charcoal adsorber system as required by Section 3.11, the following conditions shall be demonstrated. After the conditions have been demonstrated, the occurrence of painting, fire, or chemical release in any ventilation zone communicating with the spent fuel pool charcoal adsorber system shall require that the following conditions be redemonstrated, before fuel handling may continue, if operation of the spent fuel pool charcoal adsorber system is required per section 3.11

- a. The total air flow rate from the charcoal adsorbers shall be at least 75% of that measured with a complete set of new absorbers.
- b. In-place Freon testing, under ambient conditions, shall show at least 99% removal.
- c. The results of laboratory analysis on a carbon sample shall show 90% or greater radioactive methyl iodide removal when tested at least 150°F and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

See Chapter 5.0

SR 3.7.10.2

3.7.iii SR 3.7.10.1

SR 3.7.10.3

3.7.iii

- d. Flow shall be maintained through the system using either the filter or ~~bypass~~ flow path for at least 15 minutes each month.

4.11.1.2 After each replacement of a charcoal filter drawer or after any structural maintenance on the charcoal housing for the spent fuel pit charcoal adsorber system, the condition of Specification 4.11.1.1.b shall be demonstrated for the affected portion of the system.

4.11.2 Residual Heat Removal and Coolant Circulation

4.11.2.1 When the reactor is in the refueling mode and fuel is in the reactor, at least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 4 hours.

4.11.2.2 When the water level above the top of reactor vessel flange is less than 23 feet, both RHR pumps shall be verified to be operable by performing the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a.

4.11.3 Water Level - Reactor Vessel

4.11.3.1 The water level in the reactor cavity shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods in containment.

Basis

The measurement of the air flow assures that air is being withdrawn from the spent fuel pit area and passed through the adsorbers. The flow is measured prior to employing the adsorbers to establish that

See Chapter 5.0

See Chapter 3.9

~~there has been no gross change in performance since the system was last used.~~ The Freon test provides a measure of the amount of leakage from around the charcoal adsorbent.

The ability of charcoal to adsorb iodine can deteriorate as the charcoal ages and weathers. Testing the capacity of the charcoal to adsorb iodine assures that an acceptable removal efficiency under operating conditions would be obtained. The difference between the test requirement of a removal efficiency of 90% for methyl iodine and the percentage assumed in the evaluation of the fuel handling accident provides adequate safety margin for degradation of the filter after the tests.

Retesting of the spent fuel pit charcoal adsorber system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.

Testing of the air filtration systems will be tested, to the extent it can be given the configuration of the systems, in accordance with ANSI NS10-1975, "Testing of Nuclear Air-Cleaning Systems".

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

Reference:

- (1) Letter from E. J. Nelson, Rochester Gas and Electric Corporation to Dr. Peter A. Morris, U.S. Atomic Energy Commission, dated February 3, 1971

5.4 Fuel Storage Specification

5.4.1 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water.

5.4.2 The new and spent fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations. The spent fuel storage racks are divided into two regions as depicted on Figure 5.4-1. The

47.i

fuel is stored vertically in an array with sufficient center to center distance between assemblies to assure $K_{eff} \leq$

4.3.1.1.a
4.3.1.1.b

0.95 for (1) unirradiated fuel assemblies delivered prior to January 1, 1984 (Region 1-15) ~~containing no~~ ^{with initial enrichment} ~~more than 39.0 gms U-235 per axial cm,~~ ~~no greater than 3.50 weight percent U-235~~

47.ii

and (2) unirradiated fuel assemblies delivered after January 1, 1984 ~~containing~~ ^{with initial enrichment} ~~no more than 41.9 gms U-235 per axial cm,~~ ~~enrichment no greater than 4.05 weight percent U-235~~

Both cases assume unborated water used in the pool.

5.4.3 In Region 2 of the spent fuel storage racks, fuel is stored in a close packed array utilizing fixed neutron poisons in each of the stored locations. For discharged fuel assemblies to be stored in Region 2, (1) 60 days

47.iii

must have elapsed since the core reached hot shutdown prior to discharge and (2) the combination of assembly

30.17

average burnup and initial U-235 enrichment must be such that the point identified by these two parameters on Figure 5.4-2 is above the line applicable to the particular fuel assembly design, therefore assuring that $K_{eff} \leq 0.95$.



5.4.4

Cannisters containing consolidated fuel rods may be stored in either Region 1 or 2 provided that:

4.3.1.1.c

a. the average burnup and initial enrichment of the fuel assemblies from which the rods were removed satisfy the requirements of 5.4.2 and 5.4.3 above, and

47.iv

b. the average decay heat of the fuel assembly from which the rods were removed is less than 2150 BTU/hr

5.4.5

4.3.1.1.c

The requirements of 5.4.4a may be excepted for those consolidated fuel assemblies of Region RGAF2.

5.4.6

3.9.1

47.L

The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit.

Basis

The center to center spacing of Region 1 insures that $K_{eff} \leq 0.95$ for the enrichment limitations specified in 5.4.2¹, and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100².

In Region 2, $K_{eff} \leq 0.95$ is insured by the addition of fixed neutron poison (boraflex) in each of the Region 2 storage locations, and a minimum burnup requirement as a function of initial enrichment for each fuel assembly design. The 60 day cooling time requirement insures that for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100.

The two curves of Figure 5.4-2 divide the fuel assembly designs into two groups. The first group is all fuel delivered prior to January 1, 1984. This incorporates all Exxon and Westinghouse HIPAR designs used at Ginna.⁴ The second curve is for the Westinghouse Optimized Fuel Assembly design delivered to Ginna beginning in February 1984.³

The assembly average burnup is calculated using INCORE generated power sharing data and the actual plant operating history. The calculated assembly average burnup should be reduced by 10% to account for uncertainties. An uncertainty of 4% is associated with the measurement of power sharing. The additional 6% provides additional margin to bound the burnup uncertainty associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.³

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The record of these calculations shall be kept for as long as fuel assemblies remain in the pool.

The fuel storage cannisters are designed so that, normally, they can contain the equivalent number of fuel rods from two fuel assemblies in a close packed array, and can be stored in either Region 1 or Region 2 rack locations. The close packed array will insure the K_{∞} of the rack configuration containing any number of cannisters will be less than that for stored fuel assemblies at the same burnup and initial enrichment. The exception

of paragraph 5.4.5 is possible because the consolidated configuration is substantially less reactive than that of a fuel assembly. The maximum decay heat requirement will insure that local and film boiling will not occur between the close packed fuel rods if the pool temperature is maintained at or below 150°F. The decay heat of the assembly will be determined using ANS 5.1, ASB 9-2 or other acceptable substitute standards.

With the addition of the storage of consolidated fuel cannisters, the theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2x1016). However, due to limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1016 fuel assemblies.⁵

(47.7)

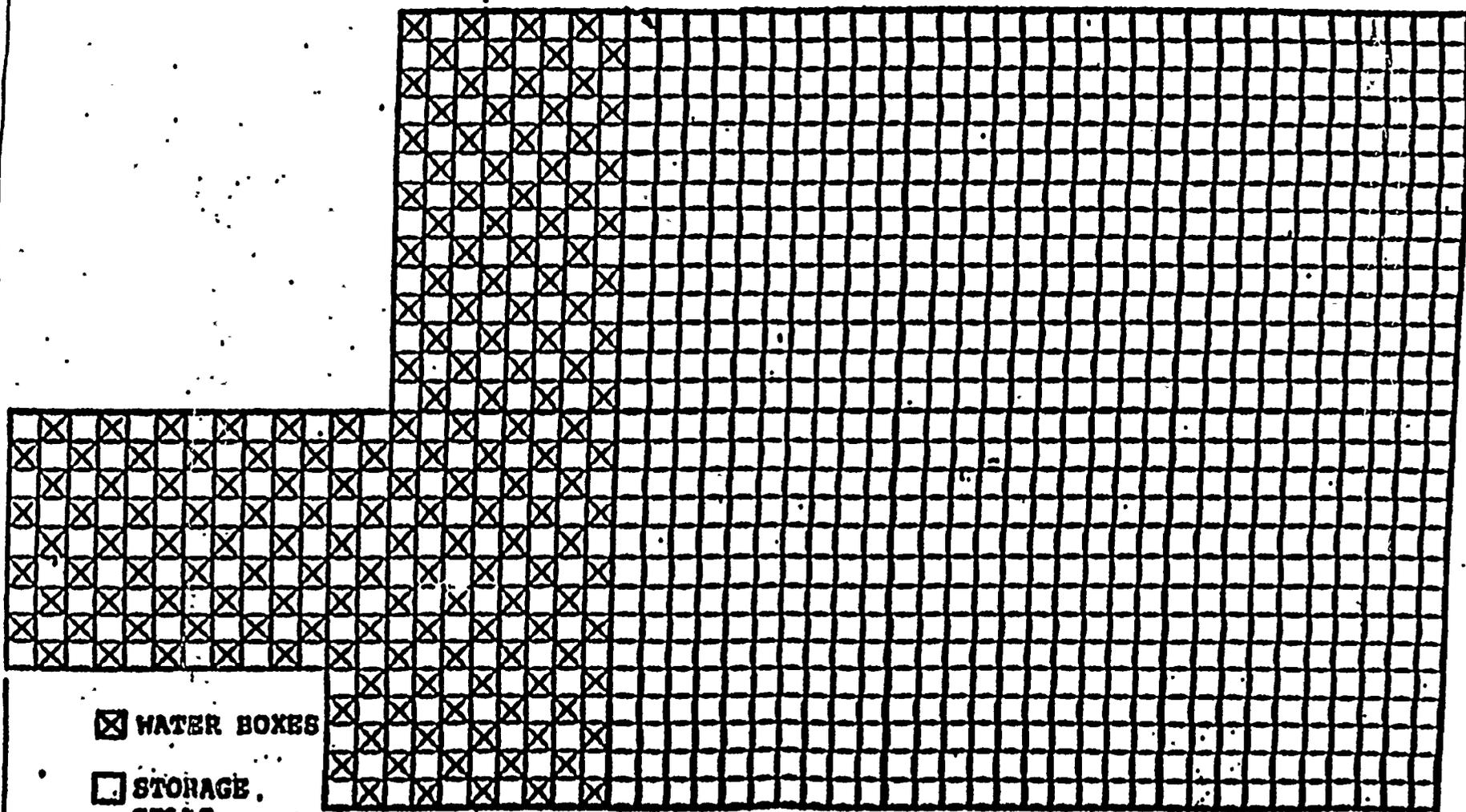
References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RGSE March 15, 1984.
5. Letter, D.M. Crutchfield to J.E. Maier, November 5, 1981.

FIG. 5.4-1

SPENT FUEL STORAGE RACKS

471



☒ WATER BOXES

☐ STORAGE CELLS

REGION 1

500 STORAGE

CAPACITY 176

REGION 2

1000 STORAGE

CAPACITY 840

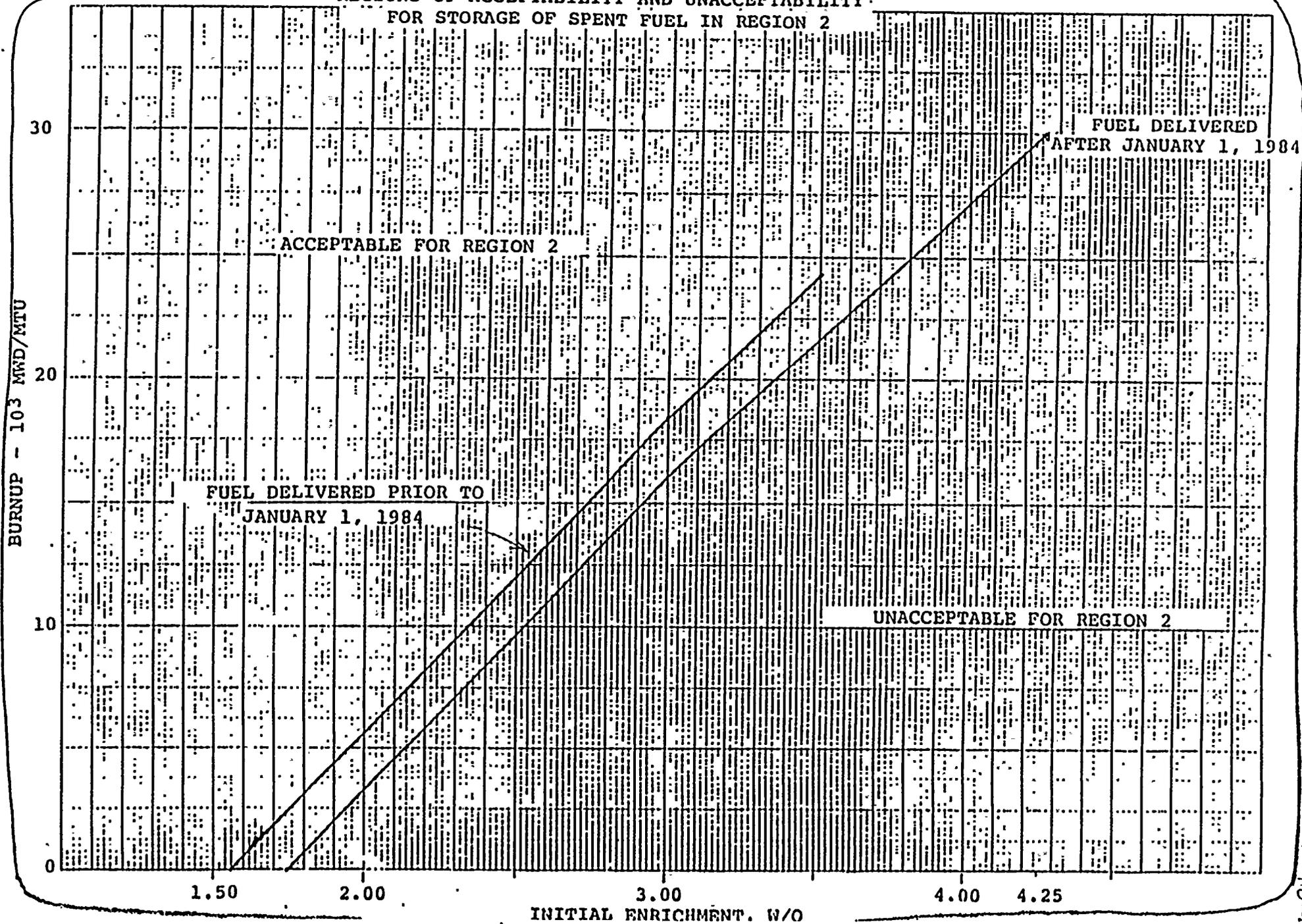
*TOTAL CAPACITY 1016 Fuel Assemblies

*Total Capacity includes provisions for storage of consolidated fuel.

3.2-35
4.0

FIGURE 5.4-2

REGIONS OF ACCEPTABILITY AND UNACCEPTABILITY
FOR STORAGE OF SPENT FUEL IN REGION 2



3.2-36-1-0-1

3.0

LIMITING CONDITION FOR OPERATION

Chapter
3.0

APPLICABILITY

3.0.1

In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, within 1 hour action shall be initiated to place the unit in at least hot shutdown within the next 6 hours (i.e., a total of seven hours), and in at least cold shutdown within the following 30 hours (i.e., a total of 37 hours) unless corrective measures are completed that permit operation under the permissible action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a mode in which the specification is not applicable. If the action statement corresponding to the Limiting Condition for Operation that was exceeded contains time limits to hot and cold shutdown that are less than those specified above, these more limiting time limits shall be applied. Exceptions to these requirements shall be stated in the individual specifications.

3.0.2

When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its preferred power source is inoperable, it may be considered operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided:

See 3.8.1

S.vi

(1) its corresponding preferred or emergency power source is operable; and (2) all of its redundant system(s), subsystems(s), train(s), component(s) and device(s) are operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within ~~1 hour~~ ^{12 hours}, the unit shall be placed in at least hot shutdown within the next 6 hours, and in at least cold shutdown within the following 30 hours. This specification is not applicable in cold shutdown or refueling modes.

CCO 3.8.1
 S.vi

See Chapter 3.6

Basis

Specification 3.0.1 delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.3.2 requires two Containment Spray Pumps to be operable and provides explicit action requirements if one spray pump is inoperable. Under the terms of Specification 3.0.1, if both of the required Containment Spray Pumps are inoperable, the unit is required to be in at least hot shutdown within the following 6 hours and in at least cold shutdown in the next 30 hours. These time limits apply because the time limits for one spray pump inoperable (6 hours to hot shutdown, wait 48 hours then 30 hours to cold shutdown) are less limiting. As a further example, Specification 3.3.1 requires each Reactor Coolant System accumulator to be operable and provides explicit action requirements if one accumulator is inoperable. Under the terms of Specification 3.0.1, if more than one accumulator is inoperable, within 1 hour action shall be initiated to place the unit in at least hot shutdown within 6 hours and cold shutdown within an additional 30 hours. The time limit of 6 hours

See Chapter
3.0

to hot shutdown and 30 hours to cold shutdown do not apply because the time limits for 1 accumulator inoperable are more limiting. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate action statement.

Specification 3.0.2 delineates what additional conditions must be satisfied to permit operation to continue, consistent with the action statements for power sources, when a preferred or emergency power source is not operable. It allows operation to be governed by the time limits of the action statement associated with the Limiting Condition for Operation for the preferred or emergency power source, not the individual action statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its preferred or emergency power source.

For example, Specification 3.7.2.1.a requires in part that two emergency diesel generators be operable. The action statement provides for a maximum out-of-service time when one emergency diesel generator is not operable. If the definition of operable were applied without consideration of Specification 3.0.2, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.2 permit the time limits for continued operation to be consistent with the action statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding

preferred power source must be operable, and all redundant systems, subsystems, trains, components, and devices must be operable, or otherwise satisfy Specification 3.0.2 (i.e., be capable of performing their design function and have at least one preferred or one emergency power source operable). If they are not satisfied, shutdown is required in accordance with this specification.

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of electrical power for the operation of plant auxiliaries.

Objective

To define those conditions of electrical power availability necessary to provide for the continuing availability of engineered safeguards.

3.7.1 Specification

3.7.1.1 When in cold shutdown or refueling, with fuel in the reactor vessel, the following conditions are to be met:

- LCO 3.8.2 a. One independent offsite power source operable, or backfeed through unit auxiliary transformer 11; and
- LCO 3.8.10 b. One train of 480-volt buses (14 and 18, or 16 and 17) operable; and
- LCO 3.8.2 c. One diesel generator operable with onsite supply of 5,000 gallons of fuel available and either buses 14 and 18, or 16 and 17, capable of being supplied from that diesel generator.
- LCO 3.8.5 d. One battery and one dc system, and at least 150 amps of battery charger capacity to the battery must be operable.
- LCO 3.8.8 e. Either 120 volt A.C. Instrument Bus 1A or 1C energized from its associated inverter.

3.7.1.2 Actions To Be Taken If Conditions of 3.7.1.1. Are Not Met:

With less than the above minimum required power source operable, immediately suspend all operations involving positive reactivity changes, core alterations, movement of

- LCO 3.8.2 Cond A/B
- LCO 3.8.5 Cond A
- LCO 3.8.8 Cond A
- LCO 3.8.10 Cond A
- LCO 3.8.3 Cond A/B/C

LCO 3.8.6 Cond A/B

17.ii

17.iv 1. Performing the surveillance requirements identified in Specifications 4.6.1.b.4 and 4.6.1.b.6 within 1 hour and at least once per 24 hours thereafter and restore the inoperable diesel generator to operable status within 7 days; OTHERWISE:

LCO 3.8.1
RA B.4

2. Reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours.

LCO 3.8.1
Cond D

8 hours 17.v

c. With one safety related 480V Bus (i.e., bus 14 or 16 or 17 or 18) de-energized, re-energize the bus within ~~1 hour~~ or reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours, unless corrective actions are complete that permit continued operation (i.e., the bus is returned to service).

LCO 3.8.9
Cond A
Cond D

d. With both independent offsite sources inoperable, both diesel generators must be operable. In addition, restore one independent offsite source within 72 hours, or reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours.

LCO 3.8.1
Cond A
Cond C
Cond D

17.vi

e. Operation above cold shutdown may continue if less than 150 amps of battery charging capacity is available to one dc system, provided at least 150 amps of battery charging capacity is available to each dc system within 2 hours. If not available, reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours.

LCO 3.8.4
Cond A
Cond B

- LCO 3.8.7 f. With either Instrument Bus 1A or 1C not energized from its associated inverter:
- RA A.1 1. Re-energize the bus within 2 hours (backup or maintenance supply), AND
- RA A.2 2. Re-energize the bus from a safety related supply (backup or inverter) within 24 hours, AND
- RA A.3 3. Re-energize the bus from its associated inverter within 72 hours, OTHERWISE
- Cond C 4. Reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours.

- LCO 3.8.7 g. With Instrument Bus 1B not energized from its associated constant voltage transformer (CVT) from MCC 1C:
- RA E.1 1. Re-energize the bus within 2 hours (maintenance supply), AND
- RA E.2 2. Re-energize the bus from its associated CVT from MCC 1C within 7 days, OTHERWISE
- Cond C 3. Reduce to a mode equal to or below hot shutdown within the next 6 hours and be in cold shutdown within the following 30 hours.

Basis for 3.7.1 and 3.7.2:

The electrical systems equipment is arranged so that no single failure can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt safeguards equipment is arranged on 4 safeguards buses. The 4160-volt equipment (none of which is safety-related), is supplied from 4 buses.

Two separate offsite sources supply station service power to the plant.

The plant auxiliary equipment is arranged electrically so that redundant safeguards loads receive power from separate sources. In the event that 1 offsite source is not available, the remaining offsite source is capable of supplying both trains of safeguards loads. Safeguards loads such as safety injection pumps, containment fans, residual heat removal pumps, and motor control centers 1C and 1D are divided between the 480-volt buses No. 14 and 16. Redundant loads including service water pumps are supplied by buses No. 18 and 17. Together these buses form the Train A and B redundant Class 1E sources.

AC power for safeguards equipment originates from both offsite and onsite sources. The operability of these power sources and associated distribution systems ensures that sufficient power will be available to supply the safety-related equipment required for (1) the

safe shutdown of the plant, and (2) the mitigation and control of accident conditions within the plant.

When the RCS is above cold shutdown, both emergency diesel generators are required to be operable. The two diesel generators have sufficient capacity to start and run all the engineered safeguards equipment at design loads. The safeguards equipment operated from one diesel generator can adequately cool the core and maintain the containment pressure within the design value for any loss of coolant incident. The minimum diesel fuel oil inventory is maintained to assure that both diesels can operate at their design ratings for 24 hours. This assures that both diesels can carry the design loads of required engineered safeguards equipment for any loss of coolant accident conditions for at least 40 hours, or for one engineered safety feature train for 80 hours.⁽¹⁾ Commercial oil supplies and trucking facilities exist to assure deliveries within 8 hours.

The offsite power source consists of separate dedicated 34.5 kv-4160 volt station service transformers served by dedicated 34.5 kv lines (12A transformer with dedicated circuit 751, or 12B transformer with dedicated circuit 767) in operable status. Either offsite source of power can supply all auxiliary loads and transfer can be accomplished within the time constraints of GDC 17. Thus, GDC 17 is explicitly met.

With fuel in the reactor vessel a minimum of one offsite source, one onsite source of AC power and one DC power train are required. The offsite power source may be provided by one of three configurations:

1. Transformer 12A served by a dedicated 34.5 kv line (circuit 751),
or
2. Transformer 12B served by a dedicated 34.5 kv line (circuit 767),
or
3. Backfeed through unit auxiliary transformer 11.

The offsite power source is the preferred source of AC power. Operability of an offsite source requires that one station service transformer served by a dedicated 34.5 kv line is operating and providing power to the unit. The emergency diesel generator provides power upon loss of the offsite source. One emergency diesel generator with 5,000 gallons of fuel can provide power to a minimum level of engineered safeguards equipment for 40 hours (the required safeguards loads at cold shutdown/refueling are significantly less than during power operation). One operable diesel fuel oil transfer pump is required to supply fuel from one of the two fuel storage tanks to the day tank of the operable diesel generator. With less than one offsite AC power source, and one onsite AC power source, one DC power train, and one battery backed instrument bus available, no operations involving positive reactivity changes, core alterations, and movement of irradiated fuel shall occur.

Battery chargers with at least 150 amps capacity shall be in service for each battery so that the batteries will always be at full charge. This ensures that adequate dc power will be available.

The plant can be safely shutdown without the use of offsite power since all vital loads (safety systems, instruments, etc.) can be supplied from the emergency diesel generators and the station batteries. Instrument Buses 1A, 1B, and 1C provide power to vital plant instrumentation. All three buses are backed up by safety related emergency supplies; bus 1A from battery 1A, bus 1C from battery 1B, and bus 1B from diesel generator 1A.

The two diesel generators, each capable of supplying safeguards loads, and the station auxiliary transformers provide four separate sources of power immediately available for operation of these loads. Thus, the power supply meets the single failure criteria.

References

- (1) UFSAR - Section 9.5.4

See Chapters
3.1, 3.4, and 3.5

	Test	Frequency
11. Service Water System	Functioning	Each Refueling Shutdown
12. Fire Protection Pump and Power Supply	Functioning	Monthly
13. Spray Additive Tank	NaOH Concent	Monthly
14. Accumulator	Boron Concentration	Bi-Monthly
15. Primary System Leakage	Evaluate	Daily

SR 2.8.14 16. Diesel Fuel Supply Fuel Inventory ~~Daily~~ 31 days

SR 3.8.2.1
28.ii.v 17. Spent Fuel Pit Boron Concentration Monthly

18. Secondary Coolant Samples Gross Activity 72 hours (2) (3)

19. Circulating Water Flood Protection Equipment Calibrate Each Refueling Shutdown

Notes:

- (1) Also required for specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.
- (2) Not required during a cold or refueling shutdown.
- (3) An isotopic analysis for I-131 equivalent activity is required at least monthly whenever the gross activity determination indicates iodine concentration greater than 10% of the allowable limit but only once per 6 months whenever the gross activity determination indicates iodine concentration below 10% of the allowable limit.
- (4) When BAST is required to be operable.

See Chapters
3.3 and 3.7

4.6 Preferred and Emergency Power Systems Periodic Tests

Applicability

Applies to periodic testing and surveillance requirements of the preferred and emergency power systems.

Objective

To verify that the preferred and emergency power systems will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

At least one diesel generator shall be demonstrated operable:

a. During cold or refueling shutdown at least once per 31 days by:

SR 3.8.2.1

33.i

- 1. Verifying the diesel starts from normal standby conditions, and attains rated voltage and frequency.

Each diesel generator shall be demonstrated operable:

b. Except during cold or refueling shutdown at least once per 31 days by:

SR 3.8.1.1

- 1. Verifying the fuel level in the day tank.

SR 3.8.3.1

- 2. Verifying a minimum oil storage of 5,000 gallons for each generator that is onsite.

SR 3.8.1.5

- 3. Verifying the fuel transfer pump can be started and transfer fuel from the storage system to the day tank.

SR 3.8.1.2

- 4. Verifying the diesel starts from normal standby conditions, and attains rated voltage and frequency.

SR 3.8.1.3

33.xi

33.xii

- 5. Verifying the generator is synchronized, loaded to at least 1950 kw but less than the 2 hour rating of 2250 kw and operates for at least 60 minutes but less than 120 minutes. (Must be performed after test on e.3)

LCO 3.8.1

33.ii

- 6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

SR 3.8.4

33.iii

c. The tests in Specification 4.6.1b will be performed prior to exceeding cold shutdown if the time since the last test exceeds 31 days.

SR 3.8.3.2

33.iv

See Chapter 5.0

d. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment.

33.v

e. At least once per ~~6~~⁽²⁴⁾ months during shutdown by:
1. Inspecting the diesel in accordance with the manufacturer's recommendations for this class of standby service.

SR 3.8.1.7

33.xii

2. Verifying the generator capability to reject a load of 295 KW without tripping.

SR 3.8.1.9

33.xii

3. Simulating a loss of offsite power in conjunction with a safety injection test signal and:

SR 3.8.1.9.a

SR 3.8.1.9.b

(a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

SR 3.8.1.9.c

(b) Verifying the diesel ^{permanently and} starts from normal standby condition on the auto-start signal, energizes the automatically connected emergency loads with the following maximum

33.vi

breaker closure times after the initial starting signal for Trains A and B not being exceeded

	A	B
Diesel plus Safety Injection	20 sec	22 sec
Pump plus RHR Pump		
All Breakers	40 sec	42 sec

SR 3.8.1.8

(c) Verifying that all diesel generator trips, except engine overspeed, low lube oil pressure, and overcrank, are automatically bypassed upon a safety injection actuation signal.

SR 3.8.1.2.
NOTE 1

4. This test may also serve to concurrently meet the requirements of 4.6.1.a and b.

4.6.2 Station Batteries

SR 3.8.6.4
SR 3.8.6.2
SR 3.8.6.1
SR 3.8.6.3

33.vii

SR 3.8.6.1
SR 3.8.6.6
SR 3.8.6.5

33.vii

- a. Every month the voltage of each cell (to the nearest 0.01 volt), the specific gravity and temperature of a pilot cell in each battery shall be measured and recorded. → 92 days
- b. Every 3 months the specific gravity of each cell, the temperature reading of every fifth cell, the height of electrolyte, and the amount of water added shall be measured and recorded.

33.ix

SR 3.8.4.1
SR 3.8.5.1

Verify battery terminal voltage
is $\geq 129V$ on float charge every
7 days.

33. x c. At each time data is recorded, new data shall be compared with old to detect signs of deterioration.

24 months d. Each battery shall be subjected to a load test within a twelve-month period from the last load test; however, to permit the load test to coincide with a scheduled refueling, the period may extend for an additional three months. The battery voltage as a function of

SR 3.8.4.2 time shall be monitored to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

SR 3.8.4.3 e. Each battery shall be subject to a discharge test at least once per 60 months. The purpose of this test is to show that the battery capacity is at least 80% of the manufacturer's recommendations. When performed,

SR 3.8.4.2 Note 1 this discharge test may substitute for the load test.

SR 3.8.4.3 f. The discharge test shall be performed annually for any battery that shows signs of degradation. Degradation

33. viii is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous discharge tests, or is below 90% of the manufacturer's rating.

4.6.3 Preferred (Offsite) Power Supplies

Each offsite power source shall be demonstrated operable:

a. At least once per 7 days by:

- SR 3.8.1.1 SR 3.8.2.i 1. Verifying nominal voltage indications on the high-voltage side of transformers 12A and 12B; and on the 4160 volt buses 12A and 12B.
- 2. Verifying 4160 volt circuit breakers 12AX or 12BX, AND 12AY or 12BY are open.

SR 3.8.9.1 3. Verifying tie breakers 52/BT16-14 and 52/BT17-18 are open when plant mode is above 200°F.

SR 3.8.1.6 b. At least once per 18 (24) months by transferring unit power supply to 4160 volt buses 12A and 12B from the normal circuit, i.e., transformer 12A for bus 12A and transformer 12B for bus 12B to the alternate circuit, i.e., transformer 12B for bus 12A and transformer 12A for bus 12B.

4.6.4 Instrument Buses

Each safety related instrument bus required to be operable, shall be demonstrated operable at least once per 7 days by:

SR 3.8.9.1
SR 3.8.10.1

1. Verifying nominal voltage indications on the Instrument Buses 1A, 1B, 1C.

SR 3.8.6.2 SR 3.8.7.1
SR 3.8.9.1 SR 3.8.7.2
SR 3.8.10.1 SR 3.8.8.1

2. Verifying proper supply breaker alignment for Instrument Buses 1A, 1B, and 1C.

SR 3.8.7.1
SR 3.8.8.1

3. Verifying proper static switch alignment for Instrument Buses 1A and 1C.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal 480V AC station service power. (1)

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are indicated by alarm. An abnormal condition in these systems can be identified without having to test the diesel generators.

Periodic tests are also specified to demonstrate that the offsite power sources will provide power for operation of equipment.

Offsite power source operability requires correct breaker alignment and indicated power availability from the two preferred power circuits, 767 and 751, to the 4160 volt buses. These requirements are met by monitoring nominal voltage indications on the high-voltage side of transformers 12A and 12B; and on the 4160 volt buses 12A and 12B.

Offsite power source independence requires separate 4160 volt circuits supplying power to the 4160 volt buses. Interlocks prevent concurrent closure of 12AX and 12BX, OR 12AY and 12BY; and surveillance is specified to ensure separation is maintained.

Instrument bus power source operability requires correct breaker alignment and indicated power availability. These requirements are met by monitoring nominal voltage indications on the buses and proper breaker alignment.

Furthermore, to assure independence between redundant Class 1E 480 volt buses 14 and 18 (Train A) and buses 16 and 17 (Train B), tie breakers 52/BT16-14 and 52/BT17-18 are required to be open when the plant mode is above 200°F. Once tie breakers are open, interlocks prevent closure when independent and redundant buses are energized.

Station batteries may deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails, and to ensure that the battery capacity is acceptable.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capability of the battery. As a check upon the effectiveness of the equalizing charge, the battery should be loaded rather heavily and the voltage monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively indicating replacement or maintenance.

The minimum permissible on-site fuel inventory, 10,000 gallons, (5,000 gallons for each generator), is sufficient for operation under loss-of-coolant accident conditions of two engineered safety features trains for 40 hours, or for one train for 80 hours, or for operation of both diesel generators at their design ratings for 24 hours. (2)

References

- (1) UFSAR, Section 8.3
- (2) UFSAR, Section 9.5.4

CTS page 3.5-2 is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

CTS page 3.5-2a is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

CTS page 3.5-4a is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

CTS page 3.5-20 is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

CTS page 3.5-20a is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

CTS page 3.5-22 is no longer contained in Attachment B, Section 3.9. The requirements on this CTS page are provided in Attachment B, Section 3.3.

3.6 Containment System

Applicability:

Applies to the integrity of reactor containment.

Objective:

To define the operating status of the reactor containment for plant operation.

Specification:

Addressed in Chapter 3.6

3.6.1 Containment Integrity

a. Except as allowed by 3.6.3, containment integrity shall not be violated unless the reactor is in the cold shutdown condition. Closed valves may be opened on an intermittent basis under administrative control.

~~b. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is greater than 2000 ppm.~~

16.x

c. Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact unless the boron concentration is greater than 2000 ppm.

LCO 3.9.1 Condition A

3.6.2 Internal Pressure

If the internal pressure exceeds 1 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within 24 hours or the reactor rendered subcritical.

Addressed in Chapter 3.6



3.8

REFUELING

Applicability

Applies to operating limitations during refueling operations

CORE ALTERATIONS and irradiated fuel assembly movement with containment

18.i

Objective

To ensure that no incident could occur during refueling operations that would affect public health and safety

Specification

3.8.1 During refueling operations the following conditions shall be satisfied.

a. Containment penetrations shall be in the following status:

LCO 3.9.3.a

i. The equipment hatch shall be in place with at least one access door closed, or the closure plate that restricts air flow from containment shall be in place,

LCO 3.9.3.b

ii. At least one access door in the personnel air lock shall be closed, and

iii. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

18.viii

1. Closed by an automatic isolation valve, blind flange, or manual valve, or or equivalent

2. Be capable of being closed by an OPERABLE automatic shutdown purge or mini-purge valve. Containment Purge and Exhaust Isolation System

18.ii

b. Radiation levels in the containment shall be monitored continuously.

LCO 3.9.2

c. Core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment and control room available whenever

18.vii
18.iii

core geometry is being changed. When core geometry is not being changed at

18.iii

least one source range neutron flux monitor shall be in service.

LCO 39.4 d.

At least one residual heat removal loop shall be in operation.*

LCO 39.1 18.iv

Immediately before reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 2000 ppm shall be maintained in the primary coolant system and checked by sampling twice each

SR 39.1 18.iv

shift every 72 hours

18.v

f. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

LCO 39.5 g.

In addition to the requirements of paragraph 3.8.1.d, while in the refueling mode with less than 23 feet of water above the top of the reactor vessel flange, two residual heat removal loops shall be operable.*

LCO 39.6 h.

During movement of fuel or control rods within the reactor vessel cavity, at least 23 feet of water shall be maintained over the top of the reactor vessel.

18.vi

* Either the preferred or the emergency power source may be inoperable for each residual heat removal loop.

Added LCO 3.9.2 RA B.2
RA C.3

18.iii

LCO 3.9.5

flange. If this condition is not met, all operations involving movement of fuel or control rods in the reactor vessel shall be suspended.

3.8.2

If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease; work shall be initiated to correct the violated conditions so that the specified limits are met; no operations which may increase the reactivity of the core shall be made.

- LCO 3.9.1, Cond A
- LCO 3.9.2, Cond A/B/C
- LCO 3.9.3, Cond A
- LCO 3.9.4, Cond A
- LCO 3.9.5, Cond A/B
- LCO 3.9.6, Cond A

18. ix

18. x

3.8.3

If the conditions of 3.8.1.d are not met, then in addition to the requirements of 3.8.2, isolate the shutdown purge and mini-purge penetrations within 4 hours.

LCO 3.9.4

LCO 3.9.5

Basis:

The equipment and general procedures to be utilized during refueling are discussed in the UFSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard

28. ii. i

New surveillance requirements:

- SR 3.9.3.1
- SR 3.9.4.1
- SR 3.9.5.1

to public health and safety. (1) Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration. The shutdown margin as indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 230,000 gallons of borated water. The boron concentration of this water at 2000 ppm boron is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold condition with all rods inserted (best estimate of 10% subcritical), and will also maintain the core subcritical even if no control rods were inserted into the reactor. (2) Periodic checks of refueling water boron concentration insure the proper shutdown margin. Communication requirements allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement. In addition to the above safeguards, interlocks are utilized during refueling to insure safe handling. An excess weight interlock is

provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time. In addition, interlocks on the auxiliary building crane will prevent the trolley from being moved over stored racks containing spent fuel.

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

The analysis⁽³⁾ for a fuel handling accident inside containment establishes acceptable offsite limiting doses following rupture of all rods of an assembly operated at peak power. No credit is taken for containment isolation or effluent filtration prior to release. Requiring closure of penetrations which provide direct access from containment atmosphere to the outside atmosphere establishes additional margin for the fuel handling accident and establishes a seismic envelope to protect against the potential consequences of seismic events during refueling. Isolation of these penetrations may be achieved by an OPERABLE shutdown purge or mini-purge valve, blind flange, or isolation valve. An OPERABLE shutdown purge or mini-purge valve is capable of being automatically isolated by R11 or R12. Penetrations which do not provide direct access from containment atmosphere to the outside atmosphere support containment integrity by either a closed system, necessary isolation valves, or a material which can provide a temporary ventilation barrier, at atmospheric pressure, for the containment penetrations during fuel movement.

References

- (1) UFSAR Sections 9.1.4.4 and 9.1.4.5
- (2) Reload Transient Safety Report, Cycle 14
- (3) UFSAR Section 15.7.3.3

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Addressed w/ Chapter 3.3

Channel Description	Check	Calibrate	Test	Remarks
1. Nuclear Power Range	S M*(3)	D(1) Q*(3)	B/W(2)(4) P(2)(5)	1) Heat balance calculation** 2) Signal to ΔT ; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset** 4) High setpoint (<109% of rated power) 5) Low setpoint (<25% of rated power)
2. Nuclear Intermediate Range	S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3. Nuclear Source Range	S(1) ↳ SR 3.9.2.1	N.A. (28.ii)	P(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4. Reactor Coolant Temperature	S	R	M(1) (2)	1) Overtemperature-Delta T 2) Overpower - Delta T
5. Reactor Coolant Flow	S	R	M	
6. Pressurizer Water Level	S	R	M	
7. Pressurizer Pressure	S	R	M	
8. 4 Kv Voltage & Frequency	N.A.	R	M	Reactor Protection circuits only
9. Rod Position Indication	S(1,2)	N.A.	M	1) With step counters 2) Log rod position indications each 4 hours when rod deviation monitor is out of service

MODE 6 REGMTS ONLY. OTHER MODES ADDRESSSED w/ CHAPTER 3.3

Addressed w/ Chapter 3.3

* By means of the movable in-core detector system.
 ** Not required during hot, cold, or refueling shutdown but as soon as possible after return to power.

Table 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

Instrument	Channel Check	Source Check	Functional Test	Channel Calibration
1. Gross Activity Monitor (Liquid)				
a. Liquid Rad Waste (R-18)	D(7)	M(4)	Q(1)	R(5)
b. Steam Generator Blowdown (R-19)	D(7)	M(4)	Q(1)	R(5)
c. Turbine Building Floor Drains (R-21)	D(7)	M(4)	Q(1)	R(5)
d. High Conductivity Waste (R-22)	D(7)	M(4)	Q(1)	R(5)
e. Containment Fan Coolers (R-16)	D(7)	M(4)	Q(2)	R(5)
f. Spent Fuel Pool Heat Exchanger A Loop (R-20A)	D(7)	M(4)	Q(2)	R(5)
g. Spent Fuel Pool Heat Exchanger B Loop (R-20B)	D(7)	M(4)	Q(2)	R(5)
Plant Ventilation				
a. Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D(7)	M	Q(1)	R(5)
b. Particulate Sampler (R-13)	W(7)	N.A.	N.A.	R(5)
c. Iodine Sampler (R-10B and R-14A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3. Containment Purge				
a. Noble Gas Activity (R-12)	D(7)	PR	Q(1)/R(1a)	R(5)
b. Particulate Sampler (R-11)	W(7)	N.A.	Q(1)/R(1a)	R(5)
c. Iodine Sampler (R-10A and R-12A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
Air Ejector Monitor (R-15 and R-15A)	D(7)	M	M(2)	R(5)
5. Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)
6. Main Steam Lines (R-31 and R-32)	M	N.A.	Q	R

See Chapter 5.0

28.v.c

See Chapter 3.3

SR 3.9.3.2

See Chapter 5.0

TABLE 4.1-5 (Continued)

TABLE NOTATIONSee Chapter
3.3

28.v.c

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm occur if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm and/or trip setpoint.
2. Power failure.

(2) The Channel Functional Test shall also demonstrate that control room alarm occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm setpoint.
2. Power failure.

(3) The Channel Calibration shall include the use of standard gas samples containing a nominal:

1. Zero volume percent oxygen; and
2. Three volume percent oxygen.

(4) This check may require the use of an external source due to high background in the sample chamber.

(5) Source used for the Channel Calibration shall be traceable to the National Bureau of Standards (NBS) or shall be obtained from suppliers (e.g. Amersham) that provide sources traceable to other officially-designated standards agencies.

(6) Flow rate for main plant ventilation exhaust and containment purge exhaust are calculated by the flow capacity of ventilation exhaust fans in service and shall be determined at the frequency specified.

(7) Applies only during releases via this pathway.

See Chapters 3.3
and 5.0

28.v.c

(ia) The test shall also demonstrate that automatic isolation of the purge valves.

See Chapter 3.7

d. Flow shall be maintained through the system using either the filter or bypass flow path for at least 15 minutes each month.

4.11.1.2 After each replacement of a charcoal filter drawer or after any structural maintenance on the charcoal housing for the spent fuel pit charcoal adsorber system, the condition of Specification 4.11.1.1.b shall be demonstrated for the affected portion of the system.

4.11.2 Residual Heat Removal and Coolant Circulation

SR 3.9.4.1
SR 3.9.3.1

38.iv

4.11.2.1 When the reactor is in the refueling mode and fuel is in the reactor, at least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

SR 3.9.4.1
SR 3.9.4.2

38.v

4.11.2.2 When the water level above the top of reactor vessel flange is less than 23 feet, both RHR pumps shall be verified to be operable by performing the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a.

4.11.3 Water Level - Reactor Vessel

SR 3.9.5.1

38.vi

4.11.3.1 The water level in the reactor cavity shall be determined to be at least its minimum required depth within 24 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods in containment.

Basis

The measurement of the air flow assures that air is being withdrawn from the spent fuel pit area and passed through the adsorbers. The flow is measured prior to employing the adsorbers to establish that

See Chapter 3.7

there has been no gross change in performance since the system was last used. The Freon test provides a measure of the amount of leakage from around the charcoal adsorbent.

The ability of charcoal to adsorb iodine can deteriorate as the charcoal ages and weathers. Testing the capacity of the charcoal to adsorb iodine assures that an acceptable removal efficiency under operating conditions would be obtained. The difference between the test requirement of a removal efficiency of 90% for methyl iodine and the percentage assumed in the evaluation of the fuel handling accident provides adequate safety margin for degradation of the filter after the tests.

Retesting of the spent fuel pit charcoal adsorber system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.

Testing of the air filtration systems will be tested, to the extent it can be given the configuration of the systems, in accordance with ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems".

~~The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.~~

Reference:

- (1) Letter from E. J. Nelson, Rochester Gas and Electric Corporation to Dr. Peter A. Morris, U.S. Atomic Energy Commission, dated February 3, 1971

5.0 DESIGN FEATURES

5.1 Site location

4.1

The K. E. Ginna Nuclear Power Plant is located ~~on property~~
~~owned by Rochester Gas and Electric Corporation at a site~~
~~site for the~~

44.i

on the south shore of Lake Ontario, approximately 16 miles east of Rochester, New York.

~~5.1.1 For the purposes of implementing Ginna Radiological Technical Specifications, and for evaluating radiological releases to the Unrestricted Area, the Unrestricted Area Boundary is assumed to coincide with the Exclusion Area Boundary. The site map shown in Figure 5.1-1 depicts the Ginna Exclusion Area Boundary (also called Unrestricted Area Boundary) location.~~

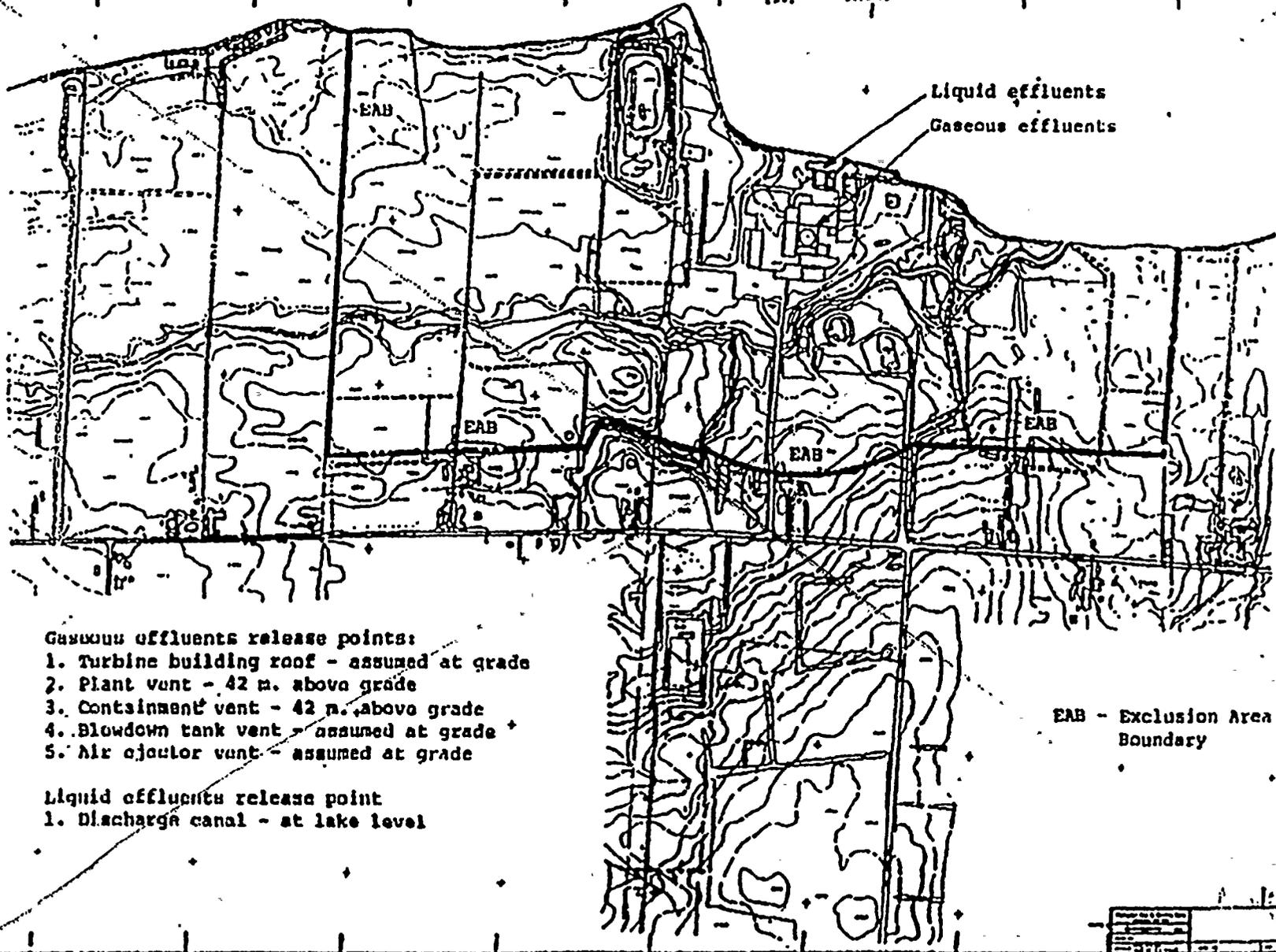
5.1.2 The site boundary shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by Rochester Gas & Electric Corporation.

44.i

GIRNA SITE MAP

Amendment No. 55

5



Gaseous effluents release points:

1. Turbine building roof - assumed at grade
2. Plant vent - 42 m. above grade
3. Containment vent - 42 m. above grade
4. Blowdown tank vent - assumed at grade
5. Air ejector vent - assumed at grade

Liquid effluents release point

1. Discharge canal - at lake level

EAB - Exclusion Area Boundary

FIGURE 5.1-1

471.1

Replace w/ UPRR - 2.3-25

Table 2.3-26
EXCLUSION AREA BOUNDARY DISTANCES

<u>Direction</u> ^a	<u>Distance</u> (m)
N	8000 ^b
NNE	8000
NE	8000
ENE	8000
E	747
ESE	640
SE	503
SSE	450
S	450
SSW	450
SW	503
WSW	915
W	945
WNW	701
NW	8000
NNW	8000

^aFrom plant toward exclusion area boundary.

^bFor calculational purposes, exclusion area boundary distances offshore were assumed to be 8000 m.

45.1

5.2 Containment Design Features

5.2.1 Reactor Containment

- a. The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The structure provides biological shielding for both normal and accident situation.
- b. The containment structure is designed for an internal pressure of 60 psig, plus the loads resulting from an earthquake producing .08g in the vertical and horizontal planes simultaneously. The containment is also structurally designed to withstand an external pressure 2.5 psi higher than the internal pressure. (1)

5.2.2 Penetrations

- a. All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts and access hatches are of the double barrier type. (2)
- b. The automatically actuated containment isolation valves are designed to close upon high pressure in the containment (setpoint no higher than 6 psig) or high radiation

5.2.1

in the containment vessel. The actuation system is designed such that no single component failure will prevent containment isolation if required.

5.2.3 Containment Systems

- a. The containment vessel has an internal spray system which is capable of providing a distributed borated water spray of at least 1200 gpm. During the initial period of spray operation, sodium hydroxide would be added to the spray water to increase the removal of iodine from the containment atmosphere. (3)
- b. The containment vessel has an internal air recirculation system which consists of four ventilation fans and air coolers capable of a total heat removal capability of 55,600 Btu/sec under conditions following a loss of coolant accident. Two of the fan cooler units are equipped with activated charcoal filters to remove volatile iodine following an accident. (4)

References:

- (1) FSAR - Section 5.1
- (2) FSAR - Section 5.1.2.7
- (3) FSAR - Section 6.4
- (4) FSAR - Section 6.3

5.3 Reactor Design Features

5.3.1 Reactor Core

4.2.1

46.i

a. The reactor core contains approximately 45 metric tons of uranium in the form of uranium dioxide pellets... The pellets are encapsulated in Zircaloy 4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies⁽¹⁾ with each fuel assembly containing 179 fuel rod locations. Fuel rod locations at any time during plant life, may consist of fuel rods clad with Zircaloy -4 or filler rods fabricated from Zircaloy -4 or stainless steel if justified by cycle-specific reload analysis. Should more than 30 rods in the core, or 10 rods in any assembly be replaced per refueling, a report describing the number of rods replaced and associated cycle-specific evaluation shall be submitted to the Commission prior to criticality. Each fuel assembly also contains 16 guide tubes and one instrumentation thimble all arranged in a 14 x 14 array to form a fuel assembly.

4.3.1.2.a

46.ii

46.iii

b. The enrichment of reload fuel shall be no more than 3.5 weight per cent U-235 for regions delivered prior to January 1, 1984 (Regions 1-15), ^{5.05}4.25 weight per cent U-235 for regions delivered after January 1, 1984, or their equivalents in terms of reactivity.

4.2.2

46.i

c. There are 29 full-length assemblies in the reactor core. Each RCC assembly contains 16 144 inch lengths of silver-indium-cadmium alloy clad with stainless steel which act as neutron absorbers when inserted into the core.⁽²⁾

Basis

The DNBRs for the reconstituted assemblies are conservatively determined by assuming the filler rods are operating at the highest power in the reconstituted fuel assembly.

(46.ii)

5.3.2

Reactor Coolant System

- a. The design of the reactor coolant system complies with the code requirements. (3)
- b. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand:
 - i. The design seismic ground acceleration, 0.08g, with stresses maintained within code allowable working stresses.
 - ii. The maximum potential seismic ground acceleration, 0.2g, acting in the horizontal and vertical directions simultaneously with no loss of function.
- c. The nominal liquid volume of the reactor coolant system, at rated operating conditions, is 6236 cubic feet.

~~5.3.2~~

References:

- (1) FSAR - Section 3.2.3
- (2) FSAR - Section 3.2.1
- (3) FSAR - Section 3.2.1
- (4) FSAR - Section 3.2.3
- (5) FSAR - Section 3.2.1 and 3.2.3
- (6) FSAR - Table 4.1.9

4b, iii

5.4 Fuel Storage
Specification

5.4.1 The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pit has a stainless steel liner to ensure against loss of water.

5.4.2 The new and spent fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations. The spent fuel storage racks are divided into two regions as depicted on Figure 5.4-1. The

47.i

fuel is stored vertically in an array with sufficient center to center distance between assemblies to assure $K_{eff} < 0.95$ for (1) unirradiated fuel assemblies delivered prior to January 1, 1984 (Region 1-15) ^{with initial enrichment} ~~containing no~~ ~~more than 39.0 gms U-235 per axial cm~~ ^{no greater than 3.50 weight percent U-235} and (2) unirradiated fuel assemblies delivered after January 1, 1984 ^{with initial enrichment} ~~containing~~ ~~no more than 41.9 gms U-235 per axial cm~~ ^{enrichment no greater than 4.05 weight percent U-235}. Both cases assume unborated water used in the pool.

4.3.1.1.a
4.2.1.1.b

47.ii

5.4.3 In Region 2 of the spent fuel storage racks, fuel is stored in a close packed array utilizing fixed neutron poisons in each of the stored locations. For discharged fuel assemblies to be stored in Region 2, (1) 60 days

must have elapsed since the core reached hot shutdown

47.iii

prior to discharge and (2) the combination of assembly average burnup and initial U-235 enrichment must be such that the point identified by these two parameters on Figure 5.4-2 is above the line applicable to the particular fuel assembly design, therefore assuring that $K_{eff} < 0.95$.

32.3

5.4.4

Cannisters containing consolidated fuel rods may be stored in either Region 1 or 2 provided that:

4.3.11.c

a. the average burnup and initial enrichment of the fuel assemblies from which the rods were removed satisfy the requirements of 5.4.2 and 5.4.3 above, and

47.iv

b. the average decay heat of the fuel assembly from which the rods were removed is less than 2150 BTU/hr

5.4.5

4.3.11.c

The requirements of 5.4.4a may be excepted for those consolidated fuel assemblies of Region RGAF2.

5.4.6

3.9.1

47.L

The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit.

Basis

The center to center spacing of Region 1 insures that $K_{eff} \leq 0.95$ for the enrichment limitations specified in 5.4.2¹, and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100².

In Region 2, $K_{eff} \leq 0.95$ is insured by the addition of fixed neutron poison (boraflex) in each of the Region 2 storage locations, and a minimum burnup requirement as a function of initial enrichment for each fuel assembly design. The 60 day cooling time requirement insures that for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100.

The two curves of Figure 5.4-2 divide the fuel assembly designs into two groups. The first group is all fuel delivered prior to January 1, 1984. This incorporates all Exxon and Westinghouse HIPAR designs used at Ginna.⁴ The second curve is for the Westinghouse Optimized Fuel Assembly design delivered to Ginna beginning in February 1984.³

The assembly average burnup is calculated using INCORE generated power sharing data and the actual plant operating history. The calculated assembly average burnup should be reduced by 10% to account for uncertainties. An uncertainty of 4% is associated with the measurement of power sharing. The additional 6% provides additional margin to bound the burnup uncertainty associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.³

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The record of these calculations shall be kept for as long as fuel assemblies remain in the pool.

The fuel storage cannisters are designed so that, normally, they can contain the equivalent number of fuel rods from two fuel assemblies in a close packed array, and can be stored in either Region 1 or Region 2 rack locations. The close packed array will insure the K_{∞} of the rack configuration containing any number of cannisters will be less than that for stored fuel assemblies at the same burnup and initial enrichment. The exception

of paragraph 5.4.5 is possible because the consolidated configuration is substantially less reactive than that of a fuel assembly. The maximum decay heat requirement will insure that local and film boiling will not occur between the close packed fuel rods if the pool temperature is maintained at or below 150°F. The decay heat of the assembly will be determined using ANS 5.1, ASB 9-2 or other acceptable substitute standards.

With the addition of the storage of consolidated fuel cannisters, the theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2x1016). However, due to limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1016 fuel assemblies.⁵

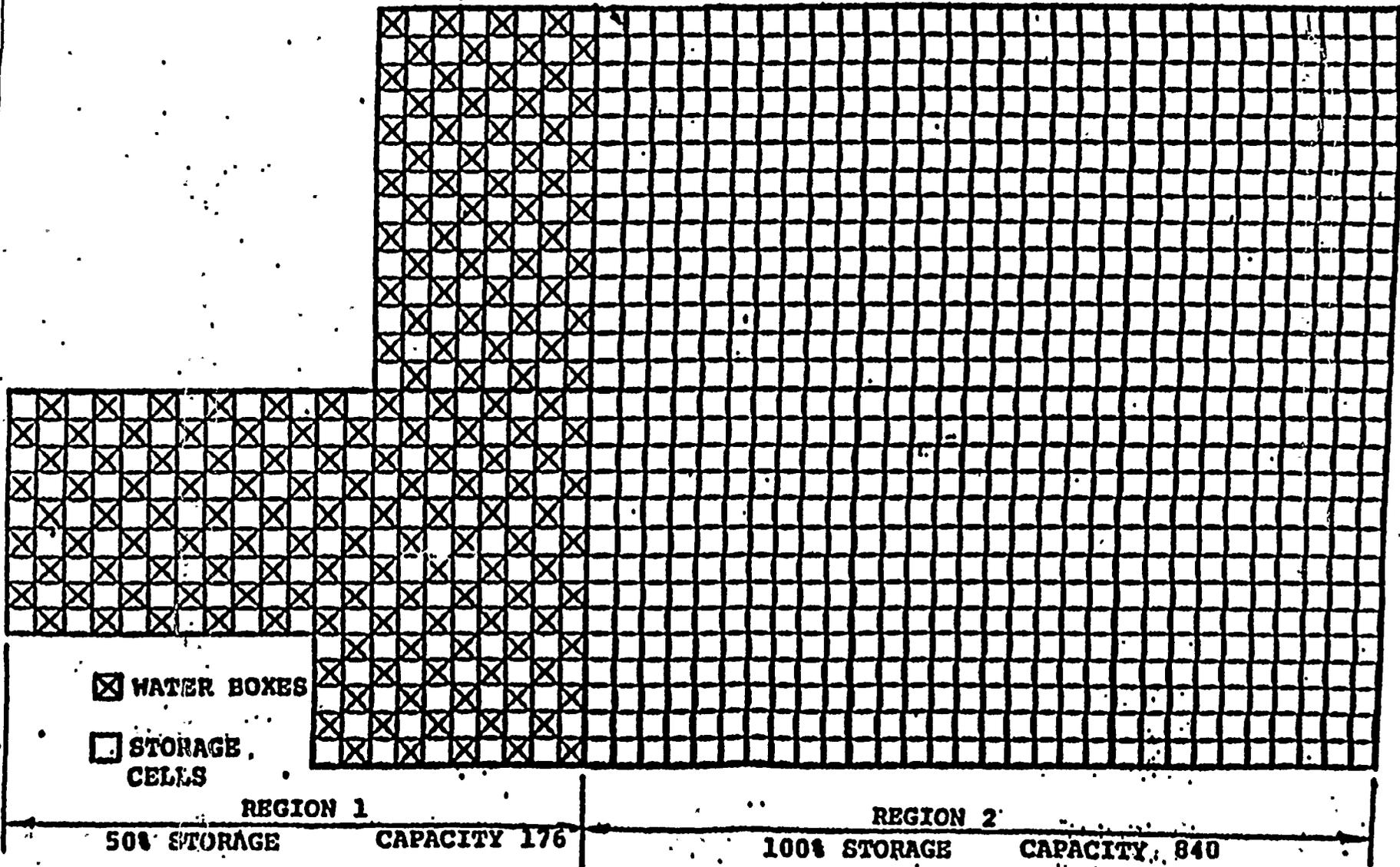
47.y

References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RG&E March 15, 1984.
5. Letter, D.M. Crutchfield to J.E. Maier, November 5, 1981.

FIG. 5.4-1

SPENT FUEL STORAGE RACKS



471



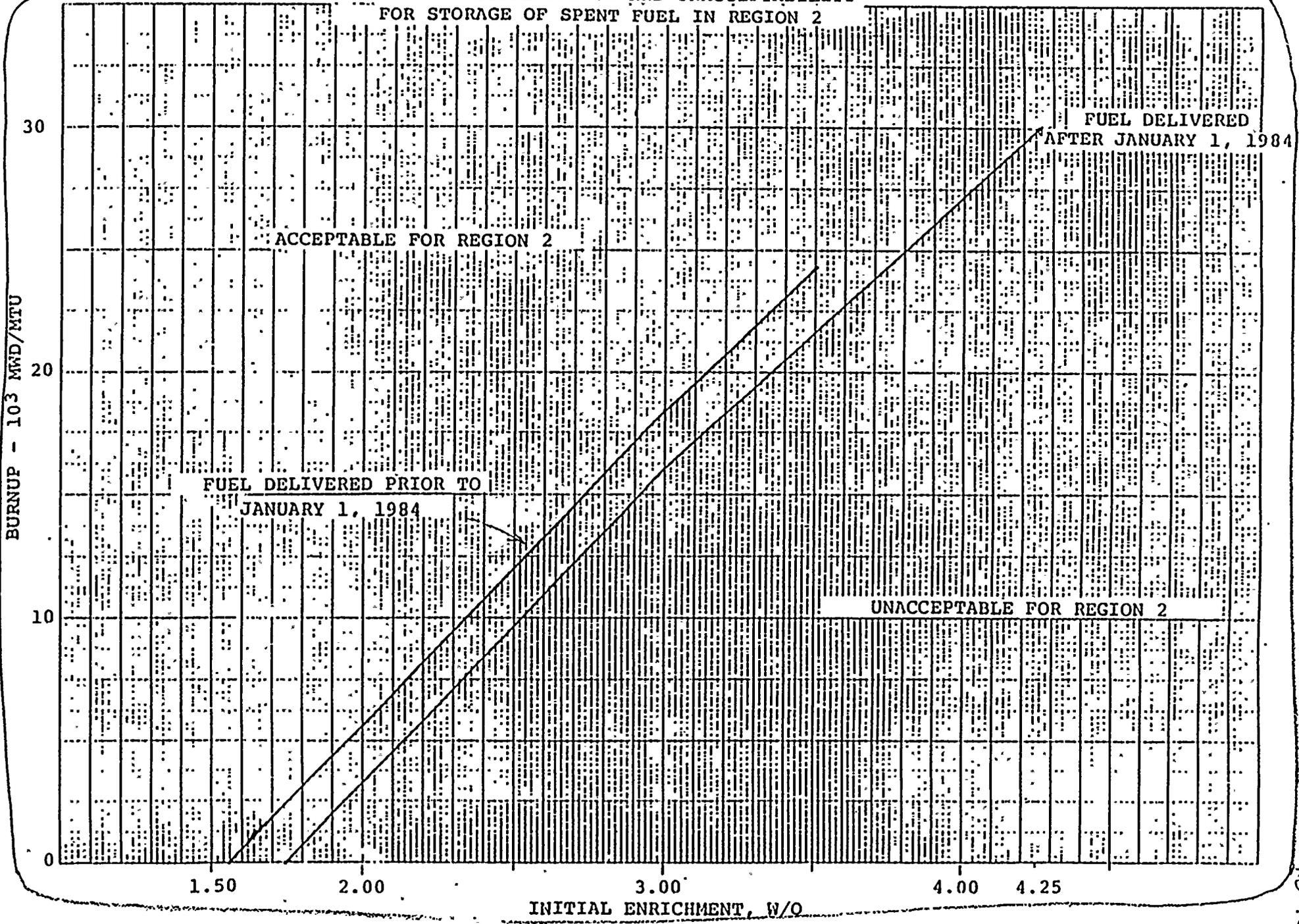
Reference to beam for UO₂ 3.7.13

*TOTAL CAPACITY 1016 Fuel Assemblies

*Total Capacity includes provisions for storage of consolidated fuel.

FIGURE 5.4-2

REGIONS OF ACCEPTABILITY AND UNACCEPTABILITY
FOR STORAGE OF SPENT FUEL IN REGION 2



BURNUP - 103 MWD/MTU

1.50

2.00

3.00

4.00

4.25

INITIAL ENRICHMENT, W/O

ACCEPTABLE FOR REGION 2

FUEL DELIVERED PRIOR TO
JANUARY 1, 1984

UNACCEPTABLE FOR REGION 2

FUEL DELIVERED
AFTER JANUARY 1, 1984

47.1

47.1

48.2

5.5 Waste Treatment Systems

5.5.1 Radioactive Liquid Waste Treatment.

The liquid waste treatment system consists of a Waste Holdup Tank, a Waste Evaporator and a mixed bed demineralize. Portions of the system may be bypassed and still meet the release limits.

5.5.2 Gaseous Radwaste Treatment

The gaseous radwaste system is designed to collect off-gas from the primary coolant system and hold for radioactive decay prior to release to the environment.

The gaseous radwaste treatment system consists of four (4) Gas Decay Tanks and two (2) gas compressors. Only one compressor and three Gas Decay Tanks are necessary to the system.

5.5.3 Ventilation Exhaust System

The ventilation exhaust is treated to reduce gaseous radioiodine and material in particulate form by passing through charcoal adsorbers and/or HEPA filters. This system has no effect on noble gas effluents. The components of the ventilation exhaust system are:

- Auxiliary Building HEPA filters
- Auxiliary Building "G" Charcoal & HEPA filters
- Auxiliary Building "A" Charcoal Adsorbers
- Containment Purge Charcoal & HEPA filters

5.5.4 Solid Radwaste System

The solid radwaste system consists of piping and valves in the Drumming Station whereby waste evaporator concentrates

are transferred into prepared drums by means of the waste evaporator feed pump. Alternatively, liquid wastes may be solidified and prepared for shipment by a contractor.

48.1

6.5

1.12

Frequency Notation

The frequency notation specified for the performance of surveillance requirements shall correspond to the intervals defined below.

<u>Notation</u>	<u>Frequency</u>
S, Each Shift	At least once per 12 hours
D, Daily	At least once per 24 hours
Twice per week	At least once per 4 days and at least twice per 7 days
W, Weekly	At least once per 7 days
B/W, Biweekly	At least once per 14 days
M, Monthly	At least once per 31 days
B/M, Bimonthly	At least once per 62 days
Q, Quarterly	At least once per 92 days
SA, Semiannually	At least once per 6 months
A, Annually	At least once per 12 months
R	At least once per 18 months
S/U	Prior to each startup
N.A.	Not Applicable
P	Prior to each startup if not done previous week
PR	Within 12 hours prior to each release

Addressed with
Chapter 1.0

(i. xii)

1.13

Offsite Dose Calculation Manual (ODCM)

5.5.1

The ODCM is a manual containing the methodology and parameters to be used for calculating the offsite

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doses due to liquid and gaseous radiological effluents, in calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.14

Process Control Program (PCP)

The PCP is a manual outlining the method for processing wet solid wastes and for solidification of liquid wastes. It shall include the process parameters and evaluation methods used to assure meeting the requirements of 10 CFR Part 71 prior to shipment of containers of radioactive waste from the site.

1.15

Solidification

Solidification shall be the conversion of radioactive wastes from liquid systems to a homogeneous solid.

1.16

Purge-Purging

Purge or purging is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confined space.

1.17

Venting

Venting is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air is not provided or required.

Addressed with
Chapter 1.0

See Chapter
3.3

3.5.3.2 When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5-3, either restore the inoperable channel(s) to operable status within 7 days, or be in at least hot shutdown within the next 12 hours.

3.5.3.3 When required by 3.5.3.1, with the number of operable accident monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table 3.5-3 either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.4 The radiation accident monitoring instrumentation channels shown in Table 3.5-6 shall be operable, whenever the reactor is at or above hot shutdown. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.5-6. Startup may commence or continue consistent with the action statement.

3.5.5 Radioactive Effluent Monitoring Instrumentation

3.5.5.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-5 shall be operable at all times, with alarm and/or trip setpoints set to insure that the limits of Specification 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

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3.5.5.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

- (i) the setpoint shall be immediately corrected without declaring the channels inoperable; or
- (ii) immediately suspend the release of effluents monitored by the effected channel; or
- (iii) declare the channel inoperable.

3.5.5.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-5. Exert best efforts to return the instruments to OPERABLE status within 31 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.6 Control Room HVAC Detection Systems

3.5.6.1 During all modes of plant operation, detection systems for chlorine gas, ammonia gas and radioactivity in the control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

See Chapter 3.2

See Chapter
3.3

chlorine, < 5 ppm
 ammonia, < 35 mg/m³
 radioactivity, particulate $\leq 1 \times 10^{-8}$ μ Ci/cc
 iodine $\leq 9 \times 10^{-9}$ μ Ci/cc
 noble gas $\leq 1 \times 10^{-5}$ μ Ci/cc

- 3.5.6.2 With one of the detection systems inoperable, within 1 hour isolate the control room HVAC air intake. Maintain the air intake isolated except for short periods, not to exceed 1 hour a day, when fresh air makeup is allowed to improve the working environment in the control room.

Basis

During plant operations, the complete instrumentation system will normally be operable. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels inoperable since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is inoperable.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

~~The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.~~

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50.

See Chapter 2.2

Control Room HVAC detection systems are designed to prevent the intake of chlorine, ammonia and radiation at concentrations which may prevent plant operators from performing their required functions. Concentrations which initiate isolation of the control room HVAC system have been established using the guidance of several established references (2-4).

The chlorine isolation setpoint is 1/3 of the toxicity limit of reference 2 but slightly greater than the short term exposure limit of reference 4. The ammonia setpoint is established at approximately 1/3 of the toxicity limit for anhydrous ammonia in reference 2 and equal to the short term exposure limit of reference 4. The setpoints for radioactivity correspond to the

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Table 3.5-5
Radioactive Effluent Monitoring Instrumentation

		<u>Minimum Channels Operable</u>	<u>Action</u>
1.	Gross Activity Monitors (Liquid)		
a.	Liquid Radwaste (R-18)	1	1
b.	Steam Generator Blowdown (R-19)	1*	2
c.	Turbine Building Floor Drains (R-21)	1	3
d.	High Conductivity Waste (R-22)	1	1
e.	Containment Fan Coolers (R-16)	1	3
f.	Spent Fuel Pool Heat Exchanger A Loop (R-20A)	1+++	3
g.	Spent Fuel Pool Heat Exchanger B Loop (R-20B)	1+++	3
2.	Plant Ventilation		
a.	Without Mini-Purge		
1.	Noble Gas Activity (R-14) (Providing Alarm and Isolation of Gas Decay Tanks)	1	4
2.	Particulate Sampler (R-13)	1	5
3.	Iodine Sampler (R-10B or R-14A***)	1	5
b.	With Mini-Purge		
1.	Noble Gas Activity (R-14)	1	4
2.	Particulate Sampler (R-13)	1	5
3.	Iodine Sampler (R-10B or R-14A***)	1	5
4.	Noble Gas Activity (R-12) or Particulate Sampler (R-11)	1++	8
3.	Shutdown Purge		
a.	Noble Gas Activity (R-12)	1+	8
b.	Particulate Sampler (R-11)	1+	8

See also Chapter 3.3

3.5-20

	<u>Minimum Channels Operable</u>	<u>Action</u>
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	c. Iodine Sampler (R-10A or R-12A***)	1+	5
4.	Air Ejector Monitor (R-15 or R-15A***)	1**	6
5.	Waste Gas System Oxygen Monitor	1	7

* Not required when Steam Generator Blowdown is being recycled (i.e. not released).
see also Chapter 3.32

+ Required only during shutdown purges and required to sample the containment stack.

++ Required to sample containment during mini-purge operation.

** Not required during Cold or Refueling Shutdown.

*** Also see Table 3.5-6.

+++ Applicable when Heat Exchanger in service.

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TABLE 3.5-5 (Continued)

Table Notation

Action 1 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases from the tank may continue for up to 14 days, provided that prior to initiating a release:

- 1. At least two independent samples of the tank's contents are analyzed, in accordance with Specification 4.12.1.1.a, and
- 2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

Action 2 - When Steam Generator Blowdown is being released (not recycled) and the number of channels operable is less than required by the Minimum Channels Operable requirements, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10^{-7} uCi/gram:

- 1. At least once per 8 hours when the concentration of the secondary coolant is > 0.01 uCi/gram dose equivalent I-131.
- 2. At least once per 24 hours when the concentration of the secondary coolant is ≤ 0.01 uCi/gram dose equivalent I-131.

Action 3 - If the number of operable channels is less than required by the Minimum Channels operable requirement, effluent releases via this pathway may continue provided that at least once per 24 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10^{-7} uCi/gm.

Action 4 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for isotopic activity within 24 hours or R14A is operable and readings are reviewed at least once per 8 hours.

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TABLE 3.5-5 (Continued)Table Notation

- Action 5 - If the number of operable channels is less than required by the Minimum Channels Operable requirements, effluent releases via this pathway may continue provided samples are continuously collected as required by Table 4.12-2 Item E with auxiliary sampling equipment.
- Action 6 - If the number of operable channels is less than required by the Minimum Channels Operable and the Secondary Activity is $\leq 1 \times 10^{-4}$ uCi/gm, effluent releases may continue via this pathway provided grab samples are analyzed for gross radioactivity (beta or gamma) at least once per 24 hours. If the secondary activity is greater than 1×10^{-4} uCi/gm, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken every 8 hours and analyzed within 24 hours.
- Action 7 - If the channel is inoperable, a sample of the gas from the in service gas decay tank shall be analyzed for oxygen content at least once every 4 hours.
- Action 8 - If the number of operable channels is less than required by the Minimum Channels Operable, or at least one containment fan cooler is not operating, within 1 hour terminate the purge.

3.9

Plant EffluentsApplicability

Applies to the controlled release of radioactive liquids and gases from the plant.

Objective

To define the conditions for release of radioactive liquid and gaseous wastes.

Specifications

19.i 3.9.1

Liquid Effluents

3.9.1.1 Concentration

3.9.1.1.a The release of radioactive liquid effluents shall be such that the concentration in the circulating water discharge does not exceed the limits specified in accordance with Appendix B, Table II, Column 2 and Notes thereto of 10CFR20. For dissolved or entrained noble gases the total activity due to dissolved or entrained noble gases shall not exceed 2×10^{-4} uCi/ml.

3.9.1.1.b If the concentration of radioactive material in the circulating water discharge exceeds the limits of 3.9.1.1.a, measures shall be initiated to restore the concentration to within those limits as soon as practicable.

3.9.1.2 Dose

3.9.1.2.a The dose or dose commitment to an individual as calculated in the ODCM from radioactive materials in liquid effluents released to unrestricted areas shall be limited:

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- (i) During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ, and
- (ii) During any calendar year to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

3.9.1.2.b Whenever the calculated dose resulting from the release of radioactive materials in liquid effluents exceeds the quarterly limits of 3.9.1.2.a(i), a Special Report shall be submitted to the Commission within thirty days which includes the following information:

- (i) Identification of the cause for exceeding the dose limit.
- (ii) Corrective actions taken and/or to be taken to reduce the releases of radioactive material in liquid effluents to assure that subsequent releases will remain within the above limits.
- (iii) The results of the radiological analyses of the nearest public drinking water source, and an evaluation of the radiological impact due to licensee releases on finished drinking water with regard to the requirements of 40 CFR 141 Safe Drinking Water Act.

3.9.1.3 Liquid Waste Treatment

3.9.1.3.a The liquid waste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge, if necessary, to assure that the cumulative dose due to liquid effluent releases when averaged

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over 31 days does not exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

3.9.1.3.b If the liquid radwaste treatment system is not operable for more than 31 days and if radioactive liquid waste is being discharged without treatment resulting in doses in excess of Specification 3.9.1.3.a, a Special Report shall be submitted to the Commission within thirty days which includes the following information:

- (i) Identification of equipment or subsystems not operable and the reasons.
- (ii) Action(s) taken to restore the inoperable equipment to operable status.
- (iii) Summary description of action(s) taken to prevent a recurrence.

3.9.2 Gaseous Wastes

3.9.2.1 Dose Rate

3.9.2.1.a The instantaneous dose rate, as calculated in the ODCM, due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

- (i) The dose rate for noble gases shall be ≤ 500 mrem/yr to the total body and ≤ 3000 mrem/yr to the skin, and
- (ii) The dose rate for all radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days shall be ≤ 1500 mrem/yr to any organ.

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3.9.2.1.b For unplanned release of gaseous wastes, compliance with 3.9.2.1.a may be determined by averaging over a 24-hour period.

3.9.2.1.c If the calculated dose rate of radioactive materials released in gaseous effluents from the site exceeds the limits of 3.9.2.1.a or 3.9.2.1.b, measures shall be initiated to restore releases to within those limits as soon as practicable.

3.9.2.1.d Compliance with 3.9.2.1.a and 3.9.2.1.b shall be determined by considering the applicable ventilation system flow rates. These flow rates shall be determined at the frequency required by Table 4.1-5.

3.9.2.2 Dose (10 CFR Part 50, Appendix I)

3.9.2.2.a The air dose, as calculated in the ODCM, due to noble gases released in gaseous effluents from the site shall be limited to the following:

- (i) During any calendar quarter to \leq 5 mrad for gamma radiation and to \leq 10 mrad for beta radiation.
- (ii) During any calendar year to \leq 10 mrad for gamma radiation and to \leq 20 mrad for beta radiation.

3.9.2.2.b The dose to an individual, as calculated in the ODCM, from radioiodine, radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than eight days released with gaseous effluents from the site shall be limited to the following:

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(i) During any calendar quarter to ≤ 7.5 mrem to any organ.

(ii) During any calendar year to ≤ 15 mrem to any organ.

3.9.2.2.c Whenever the calculated dose to an individual resulting from noble gases or from radionuclides other than noble gases exceeds the quarterly limits of 3.9.2.2.a(i) or 3.9.2.2.b(i) a Special Report shall be submitted to the Commission within thirty days which includes the following information:

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(i) Identification of the cause for exceeding the dose limit.

(ii) Corrective actions taken and/or to be taken to reduce releases of radioactive material in gaseous effluents to assure that subsequent releases will be within the above limits.

3.9.2.3 Gaseous Waste Treatment

3.9.2.3.a The gaseous radwaste treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge, if necessary, to assure that the cumulative air dose due to gaseous effluent releases to unrestricted areas when averaged over 31 days does not exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation to the maximally exposed individual.

3.9.2.3.b The appropriate portions of the ventilation exhaust system shall be used to reduce radioactive materials in gaseous waste prior to their discharge, if necessary,

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to assure that the cumulative dose due to gaseous effluent releases from the site when averaged over 31 days does not exceed 0.30 mrem to any organ.

3.9.2.3.c If the gaseous radwaste treatment system or ventilation exhaust system is inoperable for more than 31 days and if gaseous waste is being discharged without treatment resulting in doses in excess of Specifications 3.9.2.3.a or 3.9.2.3.b, a Special Report shall be submitted to the Commission within thirty days which includes the following information:

- (i) Identification of equipment or subsystems not operable and the reasons.
- (ii) Action(s) taken to restore the inoperable equipment to operable status.
- (iii) Summary description of action(s) taken to prevent a recurrence.

3.9.2.4 Dose (40 CFR Part 190)

3.9.2.4.a If the calculated dose from the release of radioactive materials from the plant in liquid or gaseous effluents exceeds twice the limits of Specifications 3.9.1.2.a, 3.9.2.2.a, or 3.9.2.2.b, a Special Report shall be submitted to the Commission within thirty days and subsequent releases shall be limited so that the dose or dose commitment to a real individual is limited to ≤ 25 mrem to the total body or any organ (except thyroid, which is limited to ≤ 75 mrem) for the calendar year that includes the release(s) covered by this report.

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This report shall include an analysis which demonstrates that radiation exposures to all real individuals from the plant are less than the 40 CFR Part 190 limits in accordance with methods set forth in the ODCM. Otherwise, the report shall request a variance from the Commission to permit releases to exceed 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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3.9.2.5 Explosive Gas Mixture

3.9.2.5.a The concentration of oxygen in each gas decay tank shall be limited to ~~≤ 2% by volume.~~

3.9.2.5.b If the concentration of oxygen in a gas decay tank is > 2% by volume but ≤ 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

3.9.2.5.c If the concentration of oxygen in a gas decay tank is > 4% by volume, immediately remove that tank from "reuse" or "in service" status and reduce the concentration of oxygen to ≤ 2% within 48 hours if such measures do not conflict with other radiological limits or procedures.

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3.9.2.6 Waste Gas Decay Tanks

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3.9.2.6.a The quantity of radioactivity contained in each waste gas decay tank shall be limited to less than or equal to 100,000 curies of noble gas (considered as Xe-133) at all times.

3.9.2.6.b If the quantity of radioactive material in any waste gas decay tank exceeds the limit of 3.9.2.6.a, immediately suspend all additions of radioactive material to the

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tank and reduce the tank contents within 48 hours if such measures do not conflict with other radiological limits or procedures.

3.9.2.7 Solid Radioactive Waste

- 3.9.2.7.a The solid radwaste system shall be used as applicable in accordance with the Process Control Program for the solidification and packaging of radioactive waste to ensure meeting the requirements of 10CFR Part 71 prior to shipment of radioactive wastes from the site.
- 3.9.2.7.b If the packaging requirements of 10 CFR Part 71 are not satisfied, suspend shipments of deficiently packaged solid radioactive wastes from the site until appropriate corrective measures have been taken.

Basis

Liquid wastes from the Radioactive Waste Disposal System are diluted in the Circulating Water System discharge prior to release to the lake.⁽¹⁾ With two pumps operating, the capacity of the Circulating Water System is approximately 400,000 gpm. Operation of a single circulating water pump reduces the nominal flow rate by about 50%. The circulating water flow under various operating conditions has been calculated from the head differential across the pumps and the manufacturer's head-capacity curves. Because of the low radioactivity levels in the circulating water discharge, the concentration of liquid radioactive effluents at this point is not measured directly. The concentration

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in the circulating water discharge is calculated from the measured concentration in the Waste Condensate Tank, the flow rate of the Waste Condensate Pumps, and the flow in the Circulating Water System. Radioactive effluents released to unrestricted areas on the basis of gross beta-gamma analysis are based on the assumption that I-129 and radium are not present. Accordingly, Appendix B, Table II, Column 2 of 10CFR20 will permit a concentration up to 1×10^{-7} uCi/ml in the circulating water discharge. Otherwise, if controlled on a radionuclide basis, the permitted discharge concentration will be in accordance with Note 1 of 10CFR20, Appendix B, Table II, Column 2. If the concentration of liquid wastes in the circulating water discharge equals the Maximum Permissible Concentration (MPC) as specified, the average concentration at the intake of the nearest public water supply at Ontario, New York, would be well below MPC. (2) Thus, these limitations provide additional assurance that the concentrations of water-borne radioactivity will result in only minimal potential public exposures within (1) Section II.A of Appendix I, 10 CFR Part 50, and (2) the limits of 10CFR Part 20.106(e).

The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air was converted to an equivalent concentration in water using ICRP Publication 2 methodology.

The Specifications which limit the dose to an individual from radioactive liquid effluents are provided to implement the requirements of Sections II.A, III.A and IV.A of 10 CFR Part 50, Appendix I. The Limiting Condition for Operation implements the guides set forth in Section II.A of 10 CFR Part 50, Appendix I. The Specifications provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of 10 CFR Part 50, Appendix I. The dose calculations in the ODCM implement the requirements in Section III.A of 10 CFR Part 50, Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on such models and data that the actual exposure of a real individual through appropriate pathways is unlikely to be substantially underestimated. Also, there is reasonable assurance that the operation of the plant will not result in waterborne radionuclide discharges which cause the potential exposure from the finished drinking water ingestion to exceed the requirements of 40CFR 141.

The requirements that the appropriate portions of the liquid radwaste treatment system be used when specified provided assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General

Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I. The limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of 10 CFR Part 50, Appendix I for liquid effluents. The cumulative maximum dose to an offsite individual from waterborne radioactive effluents is determined in order to verify that the average dose over a 31-day period is reasonably small, even if the liquid radwaste treatment system is not operated during that period. However, a cumulative dose which exceeds the stated limit does not necessarily imply that all portions of the liquid radwaste treatment system be used; certain subsystems may have only minimal effects on reducing doses.

The limit for dose rate is provided to ensure that the dose rate at any time at the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, to annual average concentrations exceeding the limits

specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, these occupancy times will be sufficiently small to compensate for any increase in the atmospheric diffusion factor above that for the site boundary.

The Specifications which limit the dose from radioactive gaseous effluents are provided to implement the requirements of Sections II.B, II.C, III.A and IV.A of 10 CFR Part 50, Appendix I. The Limiting Condition for Operation implements the guides set forth in Sections II.B and II.C of 10 CFR Part 50, Appendix I. The Specifications provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of 10 CFR Part 50, Appendix I.

The requirement that the appropriate portions of the gaseous radwaste treatment system and the ventilation exhaust treatment system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I. The limits governing the use of appropriate portions of

the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of 10 CFR Part 50, Appendix I, for gaseous effluents. The cumulative maximum dose to an offsite individual from airborne radioactive effluents is determined in order to verify that the average dose over a 31-day period is reasonably small, even in the unlikely event that the gaseous radwaste treatment or ventilation exhaust systems are not operated during that period.

However, a cumulative dose which exceeds the stated limit does not necessarily imply that all portions of the gaseous and ventilation exhaust treatment systems be used; certain subsystems may have only minimal effect on reducing doses.

The Specification on dose (40 CFR Part 190) is provided to meet the reporting requirements of 40 CFR Part 190. Since the plant is well removed from other fuel cycle facilities, it is sufficient to apply the Specification only to the plant in accordance with methods provided in the ODCM.

The Specification on explosive gas mixture is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gas decay tanks are maintained below the flammability limit of oxygen. Maintaining the concentration of oxygen below its flammability limits provides assurance that the releases

of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The waste gas decay tank curie limit is provided in order to assure that in the unlikely event of an uncontrolled release of a gas decay tank's contents, the resulting total body gamma exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

The requirement pertaining to solid radioactive waste is provided to assure that the solid radioactive waste system will be used as appropriate for the processing and packaging of solid radioactive wastes. The specification also establishes the Process Control Program which includes the process parameters and evaluation methods used to ensure meeting the requirements of 10 CFR Part 71 prior to being shipped offsite.

References

- (1) FSAR, Section 10.2
- (2) FSAR, Section 2, Appendix 2A
- (3) FSAR, Sections 2.6 and 2.7

3.13

Snubbers

(23.i)

Limiting Condition for Operation

3.13.1

With RCS conditions above cold shutdown, all safety-related

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snubbers shall be operable. This specification does not

apply to those snubbers installed on non safety-related systems if the snubber failure, and a resulting failure of the supported non safety-related system shown to be caused by that snubber failure, would have no adverse effect on any safety-related system.

(23.i)

Action

3.13.2

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to operable status and perform an engineering evaluation per Specification 4.14.1f on the supported component or declare the supported system inoperable and follow the appropriate action statement for that system.

Basis

Snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers may be replaced by rigid structural supports (bumpers) provided an analysis is performed to demonstrate that appropriate acceptance criteria are satisfied for design basis seismic and pipe break events and provided that the bumpers are inspected periodically in a manner appropriate for rigid structural supports.

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3.15 Overpressure Protection System

Applicability

Applies whenever the temperature of one or more of the RCS cold legs is $\leq 330^{\circ}\text{F}$, or the Residual Heat Removal System is in operation.

Objective

To prevent overpressurization of the reactor coolant system and the residual heat removal system.

Specification

3.15.1 Except during secondary side hydrostatic tests in which RCS pressure is to be raised above the PORV setpoint, at least one of the following over-pressure protection systems shall be operable:

- a. Two pressurizer power operated relief valves (PORVs) with a lift setting of ≤ 424 psig, or
- b. A reactor coolant system vent of ≥ 1.1 square inches.

3.15.1.1 With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the RCS through a 1.1 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.2 With both PORVs inoperable, depressurize and vent the RCS through a 1.1 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

5.6.4

3.15.1.3 Use of the overpressure protection system to mitigate an RCS or RHRS pressure transient shall be reported in accordance with 6.9/2.

(25.v)

Basis

An RCS vent opening of greater than 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq 330^{\circ}\text{F}$. This relief capacity will

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3.16 Radiological Environmental Monitoring

Applicability

Applies to routine testing of the plant environs.

Objective

To establish a program which will assure recognition of changes in radioactivity or exposure pathways in the environs.

Specification

3.16.1 Monitoring Program

3.16.1.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.16-1 at the locations given in the ODCM.

3.16.1.2 If the radiological environmental monitoring program is not conducted as specified in Table 3.16-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal availability, or to malfunction of automatic sampling equipment. If the latter, efforts shall be made to complete corrective action prior to the end of the next sampling period.)

3.16.1.3 If the level of radioactivity in an environmental sampling medium at one or more of the locations specified in the ODCM exceeds the reporting levels of Table

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6.9-2 when averaged over any calendar quarter, a Special Report shall be submitted to the Commission within thirty days which includes an evaluation of any release conditions, environmental factors or other aspects which caused the reporting levels of Table 6.9-2 to be exceeded.

When more than one of the radionuclides in Table 6.9-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in Table 6.9-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is greater than the calendar year limit of Specifications 3.9.1.2.a or 3.9.2.2.b. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3.16.1.4 If milk or fresh leafy vegetable samples are unavailable for more than one sample period from one or more of the sampling locations indicated by the ODCM, a discussion shall be included in the Radioactive Effluent Release Report which identifies the cause of the unavailability of samples and identifies locations for

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obtaining replacement samples. If a milk or leafy vegetable sample location becomes unavailable, the locations from which samples were unavailable may then be deleted from the ODCM, provided that comparable locations are added to the environmental monitoring program.

3.16.2 Land Use Census

3.16.2.1 A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles.

3.16.2.2 An onsite garden located in the meteorological sector having the highest historical D/Q may be used for broad leaf vegetation sampling in lieu of a garden census; otherwise the land use census shall also identify the location of the nearest garden of greater than 500 square feet in each of the 16 meteorological sectors within a distance of five miles. D/Q shall be determined in accordance with methods described in the ODCM.

3.16.2.3 If a land use census identifies a location(s) which yields a calculated dose or dose commitment greater than that of the maximally exposed individual currently being calculated in Specification 3.12.2.2, the new identified location(s) shall be reported in the Semi-annual Radioactive Release Report.

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3.16.2.4 If a land use census identifies a milk location(s) which yields a calculated dose or dose commitment greater than that at a location from which samples are currently being obtained in accordance with Specification 3.16.1, the new identified location(s) shall be reported in the Semiannual Radioactive Release Report. The new location shall be added to the radiological environmental monitoring program within thirty days, if possible. The milk location having the lowest calculated dose or dose commitment may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

3.16.3 Interlaboratory Comparison Program

3.16.3.1 Analyses shall be performed on applicable radioactive environmental samples supplied as part of an interlaboratory comparison program which has been approved by NRC, if such a program exists.

3.16.3.2 If analyses are not performed as required above, report the corrective actions taken to prevent a recurrence in the Annual Radiological Environmental Operating Report.

26.iii

Basis

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting

from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least three years. Following this period, program changes may be initiated based on operational experience. The detection capabilities required by Table 4.10-1 are state-of-the-art for routine environmental measurements in industrial laboratories. Lower limits of detection (LLDs) are intended as a priori (before-the-fact) limits, and analyses will be conducted in such a manner that the stated LLDs will be achieved under routine conditions. The land use census requirement is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. A garden census is not required if an onsite garden is located in the meteorological sector having the highest historical D/Q is used for broad leaf vegetation sampling. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid. Only samples with radioactivity levels comparable to levels in environmental samples need be analyzed.

TABLE 3.16-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
1. AIRBORNE			
a. Radioiodine	2 indicator 2 control	Continuous operation of sampler with sample collection at least once per 10 days.	Radioiodine canister. Analyze within 7 days of collection of I-131
b. Particulates	7 indicator 5 control	Same as above.	Particulate sampler. Analyze for gross beta radioactivity \geq 24 hours following filter change. Perform gamma isotopic analysis on each sample for which gross beta activity is > 10 times the mean of offsite samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION	18 indicator 10 control 11 placed greater than 5 miles from plant site	TLDs at least quarterly.	Gamma dose quarterly.

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3.16-1

TABLE 3.16-1 (CONTINUED)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
3. WATERBORNE			
a. Surface	1 control (Russell Station) 1 indicator (Condenser Water Discharge)	Composite* sample collected over a period of \leq 31 days.	Gross beta and gamma isotopic analysis of each composite sample. Tritium analysis of one composite sample at least once per 92 days.
b. Drinking	1 indicator (Ontario Water District Intake)	Same as above.	Same as above.

*Composite sample to be collected by collecting an aliquot at intervals not exceeding 2 hours.

26.1

Amendment No. 2A

TABLE 3.16-1 (CONTINUED)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

26.ii

<u>Exposure Pathway and/or Sample</u>	<u>Number of Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
4. INGESTION			
a. Milk	1 control 3 indicator June thru October each of 3 farms	At least once per 15 days.	Gamma isotopic and I-131 analysis of each sample.
	1 control 1 indicator November thru May one of the farms	At least once per 31 days.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	4 control 4 indicator (Off shore at Ginna)	Twice during fishing season including at least four species.	Gamma isotopic analysis on edible portions of each sample.
c. Food Products	1 control 2 indicator (On site)	Annual at time of harvest. Sample from two of the following: 1. apples 2. cherries 3. grapes	Gamma isotopic analysis on edible portion of sample.
	1 control 2 indicator (On site garden or nearest offsite garden within 5 miles in the highest D/Q meteorological sector)	At time of harvest. One sample of: 1. broad leaf vegetation 2. other vegetable	Gamma isotopic analysis on edible portions of each sample.

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See Chapter 3.3

4.0 SURVEILLANCE REQUIREMENTS

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 Operational Safety Review

Applicability:

Applies to items directly related to safety limits and limiting conditions for operation.

Objective:

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

4.1.1 Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.

4.1.2 Equipment and sampling tests shall be conducted as specified in Table 4.1-2 and 4.1-4.

4.1.3 Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check and channel calibration operations at the frequencies shown in Table 4.1-3.

~~4.1.4 Each radioactive effluent monitoring instrumentation channel shall be demonstrated operable by performing the channel check, source check, channel functional test, and channel calibration at the frequency shown in Table 4.1-5.~~

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See Chapter 3.3

Basis:

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor and steam system instrumentation.

Control Room procedures require a check of the Radiation Monitoring System (RMS) panel meters and strip chart recorders for proper readout once each shift. A daily surveillance log is also maintained in the Control Room for manual entry of RMS readouts, and is independently reviewed by Health Physics supervision at least weekly.

~~A radiation monitor downscale failure will result in a conspicuous visual indication on the RMS panel (no audible alarm). Radiation monitor control switches are spring-returned to the "operate" mode after being turned to any other test or check mode. Therefore, together with the design features of the RMS, plant surveillance procedures ensure the continued availability of each radiation monitor to perform its intended function.~~

TABLE 4.1-1 (Continued)

Channel Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With rod position indication 2) Log rod position indications each 4 hours when rod deviation monitor is out of service
11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Storage Tank Level	D	R	N.A.	Note 4
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
19. Boric Acid Control	N.A.	R	N.A.	
20. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
24. Accumulator Level and Pressure	S	R	N.A.	

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See Chapter 2.3

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TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
25. Containment Pressure	S	R	M	Narrow range containment pressure (-3.0, +3 psig) excluded
26. Steam Generator Pressure	S	R	M	
27. Turbine First Stage Pressure	S	R	M	
28. Emergency Plan Radiation Instruments	M	R	M	See Chapter 3.3
29. Environmental Monitors	M	NA	NA	
30. Loss of Voltage/Degraded Voltage 480 Volt Safeguards Bus	NA	R	M	
31. Trip of Main Feedwater Pumps	NA	NA	R	
32. Steam Flow	S	R	M	
33. T _{svr}	S	R	M	
34. Chlorine Detector, Control Room Air Intake	NA	R	M	See Chapter 3.3
35. Ammonia Detector, Control Room Air Intake	NA	R	M	
36. Radiation Detectors, Control Room Air Intake	NA	R	M	
37. Reactor Vessel Level Indication System	M	R	NA	
38a. Trip Breaker Logic Channel Testing	NA	NA	M	Notes 1, 2 and 3
38b. Trip Breaker Logic Channel Testing	NA	NA	R	Note 1

28.1.1

4.1-7

28.v.b

Table 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

	<u>Instrument</u>	<u>Channel Check</u>	<u>Source Check</u>	<u>Functional Test</u>	<u>Channel Calibration</u>
1.	Gross Activity Monitor (Liquid)				
a.	Liquid Rad Waste (R-18)	D(7)	M(4)	Q(1)	R(5)
b.	Steam Generator Blowdown (R-19)	D(7)	M(4)	Q(1)	R(5)
c.	Turbine Building Floor Drains (R-21)	D(7)	M(4)	Q(1)	R(5)
d.	High Conductivity Waste (R-22)	D(7)	M(4)	Q(1)	R(5)
e.	Containment Fan Coolers (R-16)	D(7)	M(4)	Q(2)	R(5)
f.	Spent Fuel Pool Heat Exchanger A Loop (R-20A)	D(7)	M(4)	Q(2)	R(5)
g.	Spent Fuel Pool Heat Exchanger B Loop (R-20B)	D(7)	M(4)	Q(2)	R(5)
	Plant Ventilation				
a.	Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D(7)	M	Q(1)	R(5)
b.	Particulate Sampler (R-13)	W(7)	N.A.	N.A.	R(5)
c.	Iodine Sampler (R-10B and R-14A)	W(7)	N.A.	M	R(5)
d.	Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3.	Containment Purge				
a.	Noble Gas Activity (R-12)	D(7)	PR	Q(1)	R(5)
b.	Particulate Sampler (R-11)	W(7)	N.A.	Q(1)	R(5)
c.	Iodine Sampler (R-10A and R-12A)	W(7)	N.A.	M	R(5)
d.	Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
	Air Ejector Monitor (R-15 and R-15A)	D(7)	M	M(2)	R(5)
5.	Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)
6.	Main Steam Lines (R-31 and R-32)	M	N.A.	Q	R

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TABLE 4.1-5 (Continued)

TABLE NOTATION

- (1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm occur if any of the following conditions exist:
 - 1. Instrument indicates measured levels above the alarm and/or trip setpoint.
 - 2. Power failure.
- (2) The Channel Functional Test shall also demonstrate that control room alarm occurs if any of the following conditions exist:
 - 1. Instrument indicates measured levels above the alarm setpoint.
 - 2. Power failure.
- (3) The Channel Calibration shall include the use of standard gas samples containing a nominal:
 - 1. Zero volume percent oxygen; and
 - 2. Three volume percent oxygen.
- (4) This check may require the use of an external source due to high background in the sample chamber.
- (5) Source used for the Channel Calibration shall be traceable to the National Bureau of Standards (NBS) or shall be obtained from suppliers (e.g. Amersham) that provide sources traceable to other officially-designated standards agencies.
- (6) Flow rate for main plant ventilation exhaust and containment purge exhaust are calculated by the flow capacity of ventilation exhaust fans in service and shall be determined at the frequency specified.
- (7) Applies only during releases via this pathway.

4.2 Inservice Inspection

Applicability

Applies to the inservice inspection of Quality Groups A, B, and C Components, High Energy Piping Outside of Containment, Snubbers and Steam Generator tubes. It also applies to inservice pump and valve testing.

29.i

Objectives

To provide assurance of the continuing structural and operational integrity of the structures, components and systems in accordance with the requirements of 10 CFR 50.55a(g).

Specification

4.2.1 The inservice inspection program for Quality Groups A, B, and

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5.5.8

C Components, High Energy Piping Outside of Containment,

Snubbers and Steam Generator tubes shall be in accordance with

Appendix B of the Ginna Station Quality Assurance Manual.

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This inservice pump and valve testing program shall be in accordance with Appendix C of the Ginna Station Quality Assurance Manual. These inservice inspection programs shall

define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, which are applicable for the forty month period of the ten year inspection interval. The programs' ten year inspection interval shall be based on the following commencing dates.

The Nelson Policy Manual.

4.2.1.1 The inspection interval for Quality Group A components shall be ten year intervals of service commencing on January 1, 1970.

4.2.1.2 The inspection intervals for Quality Group B and C Components shall be ten year intervals of service commencing with May 1, 1973, January 1, 1980, 1990 and 2000, respectively.

4.2.1.3 The inspection intervals for the High Energy Piping Outside of Containment shall be ten year intervals of service commencing May 1, 1973, January 1, 1980, 1990 and 2000, respectively. The inspection program during each third of the first inspection interval provides for examination of all welds at design basis break locations and one-third of all welds at locations where a weld failure would result in unacceptable consequences. During each succeeding inspection interval, the program shall provide for an examination of each of the design basis break location welds, and each of the welds at locations where a weld failure would result in unacceptable consequences.

4.2.1.4 The inspection intervals for Steam Generator Tubes shall be specified in the "Inservice Inspection Program" for the applicable forty month period commencing with May 1, 1973.

5.5.8.a

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4.2.1.4.a Steam generator tubes that have imperfections greater than 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.

5.5.8.b

4.2.1.4.b Steam generator sleeves that have imperfections greater than 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.8.c

4.2.1.5 Inservice Inspection of ASME Code Class 1, Class 2 and Class 3 components (Quality Groups A, B, and C) shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

29.i- 4.2.1.6 The inspection interval for the Inservice Pump and Valve Testing Program shall be ten year intervals commencing with January 1, 1981, 1990 and 2000.

4.2.1.7 The inspection intervals for Snubbers shall be as defined in Specification 4.14.

Basis

The inservice inspection program provides assurance for the continued structural integrity of the structures, components and systems of Ginna Station. The programs comply with the ASME Boiler and Vessel Code Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components" as practicable, with due consideration to the design and physical access of the structures, components and systems as manufactured and constructed. This compliance will constitute an acceptable basis for satisfying the requirements of General Design Criterion 32, Appendix A of 10 CFR Part 50 and the requirements of Section 50.55a, paragraph g of 10 CFR Part 50.

The repair criteria of 4.2.1.4.a and 4.2.1.4.b are based on the requirements of USNRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" as implemented by RG&E (Reference 1). This guide describes a method acceptable to the NRC staff for establishing the limiting safe conditions of tube degradation of steam generator tubing. The repair criteria is based on structural allowances, an allowance for eddy current measurement error and an allowance for degradation during the operating period. These allowances are added together to determine the repair criteria which is typically 40% for steam generator tubes. Based on calculations the appropriate sleeve plugging limit is a 42% thru wall defect. In order to allow for conservatism, a 30% plugging limit for sleeves will be utilized.

Reference 1: "Steam Generator Rapid Sleaving Program Design Verification Report", R.E. Ginna Nuclear Power Plant, August 1982.

~~Amendment No. 35~~

~~Correction letter of July 3, 1990 4.2-4~~

4.4.3 Recirculation Heat Removal Systems

4.4.3.1 Test

5.5.2

- a. The portion of the residual heat removal system that is outside the containment shall ~~either~~ be tested by use in normal operation or hydrostatically tested at 350 psig at the interval specified in 4.4.3.4.
- b. Suction piping from containment sump B to the reactor coolant drain tank pump and the discharge piping from the pumps to the residual heat removal system shall be hydrostatically tested at no less than 100 psig at the interval specified in 4.4.3.4.

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- c. Visual inspection shall be made for excessive leakage from components of the system. Any significant leakage shall be measured by collection and weighing or by an equivalent method.

4.4.3.2 Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.4.3.3 Correction Action

- a. Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.4.3.2.
- b. If repairs are not completed within 24 hours, the reactor shall be shut down and depressurized until repairs are effected and the acceptance criterion of 4.4.3.2 is satisfied.

4.4.3.4 Test Frequency

Tests of the recirculation heat removal system shall be conducted at intervals not to exceed 12 months.

5.5.6 4.4.4 Tendon Stress Surveillance

4.4.4.1 Inspection for Broken Wire

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- a. Fourteen specific tendons, equally spaced around the

containment shall be inspected periodically for the presence of broken wires.

- b. The inspection intervals, measured from the date of the initial structural test, shall be as follows:
- 6 months
 - 1 year
 - 3 years
 - 8 years and 5 years intervals thereafter.
- c. The acceptance criteria for the inspection are that no more than a total of 38 wires (in 14 tendons) are broken and that not more than 6 broken wires exist in any one tendon. If more than 38 broken wires are found, all tendons shall be inspected. If inspection reveals more than 5% of the total wires broken, the reactor shall be shut down and depressurized.
- d. If more than 20 wires (in 14 tendons) have been broken since the last inspection, all tendons shall be inspected. If inspection reveals more than 5% of the total wires broken, the reactor shall be shut down and depressurized.
- e. If as many as 6 broken wires are found in any one tendon, four immediately adjacent tendons (two on each side of

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the tendon containing 6 broken wires) shall be inspected. The accepted criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

- a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.
- b. Before reseating, a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each containment isolation valve shall be demonstrated to be OPERABLE in accordance with the Ginna Station Pump and Valve Test program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.4.6.2 The response time of each containment isolation valve shall be demonstrated to be within its limit at least once per 18 months. The response time includes only the valve travel time for those valves which the safety analysis assumptions take credit for a change in valve position in response to a containment isolation signal.

Addressed with Chapter 3.6

pressure, 350 psig, achieved either by normal system operation or by hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return lines and the reactor coolant drain tank piping connections to the residual heat removal system of 100 psig gives an adequate margin over the highest pressure within the lines after a design basis accident. (4)

A recirculation system leakage of 2 gal./hr will limit offsite exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident. The dose calculated as a result of this leakage is 7.7 mr for a 2-hr exposure at the site boundary. (5)

In case of failure to meet the acceptance criteria for leakage from the residual heat removal system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor. The times allowed for repairs are consistent with the times developed in Specification 3.3.

The tendon surveillance program is based on assuring that, on the average, the load-carrying capability of the tendons is maintained at approximately 95% design.

This is consistent with the design criteria for the tendons, which allow for uniform capacity reduction of 0.95 and which contemplate that a small fraction of the individual wires 0.03--0.5% may break during tensioning. (6)

Periodic visual inspection is the method to be used to determine loss of load-carrying capability because of wire breakage. Since the tendon is under a stress of approximately 144,000 psi, should a wire break, the button head will rise above the top anchor head where it can be readily observed. Assuming that 38 broken wires are observed in 14 tendons (90 wires per tendon), which corresponds to a mean breakage of 3% (97% design load-carrying capability), it can be stated with 95% confidence that the fraction of broken wires in the total containment is between 2.1 and 4.0%. This is based on reliability tables developed by North American Aviation, (7) for statistical situations that can be represented by a Poisson distribution. A condition for fitting a Poisson distribution is that the possibility of wire breakage is constant and small. The specification relating to as many as 6 broken wires in one tendon (6.6%) provides that the assumption of a constant probability of occurrence is not significantly violated. The design load can be carried even if three adjacent tendons fail completely. (8) The specification has the purpose of alerting against possible deterioration at any time in the plant operating 1st time.

The pre-stress confirmation test provides a direct measure of the load-carrying capability of the tendon.

If the surveillance program indicates by extensive wire breakage or tendon stress relation that the pre-stressing tendons are not behaving as expected, the situation will be evaluated immediately. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. The containment is provided with two readily removable tendons that might be useful to such a study. In addition, there are 40 tendons, each containing a removable wire which will be used to monitor for possible corrosion effects.

Operability of the containment isolation boundaries ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Performance of cycling tests and verification of isolation times associated with automatic containment isolation valves is covered by the Pump and Valve Test Program. Compliance with Appendix J to 10 CFR 50 is addressed under local leak testing requirements.

References:

- (1) UFSAR Section 3.1.2.2.7
- (2) UFSAR Section 6.2.6.1
- (3) UFSAR Section 15.6.4.3
- (4) UFSAR Section 6.3.3.8
- (5) UFSAR Table 15.6-9
- (6) FSAR Page 5.1.2-28
- (7) North-American-Rockwell Report 550-x-32, Autonetics Reliability Handbook, February 1963.
- (8) FSAR Page 5.1.2-28

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of the spray additive valves closed, each valve will be opened and closed by operator action. This test shall be performed prior to startup if the time since the last test exceeds one month.

c. The accumulator check valves shall be checked for operability during each refueling shutdown.

4.5.2.3 Air Filtration System

4.5.2.3.1

At least once every 18 months or after every 720 hours of charcoal filtration system operation since the last test, or following painting, fire or chemical release in any ventilation zone communicating with the system, the post accident

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5.5.10.a

charcoal system shall have the following conditions demonstrated:

- a. The pressure drop across the charcoal adsorber bank is less than 3 inches of water at design flow rate ($\pm 10\%$).
- b. In place Freon testing, under ambient conditions, shall show at least 99% removal.

32.v

c. The iodine removal efficiency of at least one charcoal filter cell shall be measured. The filter cell to be tested shall be selected randomly from those cells with the longest in-bank residence time. The minimum

32.v

acceptable value for filter efficiency is 90% for removal of methyl iodide when tested at at least 286°F

32.v

and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

32.v

4.5.2.3.2 After each replacement of a charcoal drawer or after any structural maintenance on the housing for the post accident charcoal system, the condition of Specification 4.5.2.3.1.b shall be demonstrated for the affected portion of the system.

32.v

4.5.2.3.3 At least every 18 months or following painting, fire, or chemical release in any ventilation zone communicating with the system, the containment recirculation system shall have the following conditions demonstrated.

32.v

5.5.10.b a. The pressure drop across the HEPA filter bank is less than 3 inches of water at design flow rate ($\pm 10\%$).

32.v

32.v

b. In place ~~thermally generated~~ DOP testing of the HEPA filters shall show at least 99% removal.

4.5.2.3.4 After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on a housing for the containment recirculation system, the condition of Specification 4.5.2.3.3.b shall be demonstrated for the affected portion of the system.

32.v

4.5.2.3.5 Except during cold or refueling shutdowns the post accident charcoal filter isolation valves shall be tested at intervals not greater than one month to verify operability and proper orientation and flow shall be maintained through the system for at least 15 minutes. The test shall be performed prior to startup if the time since the last test exceeds 1 month.

Addressed w/
Chapter 3.6

4.5.2.3.6

At least once every 18 months or after every 720 hours of charcoal filtration system operation since the last test, or following painting, fire or chemical release in any ventilation zone communicating with the system, the control room emergency air treatment system shall have the following conditions demonstrated.

32.v

5.5.10.c

32.vi

32.v

a. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6" of water at design flow rate ($\pm 10\%$).

b. In place Freon testing, under ambient conditions, shall show at least 99% removal.

32.v

c. In place ~~thermally generated~~ DCP testing of the HEPA filters shall show at least 99% removal.

d. The results of laboratory analysis on a carbon sample shall show 90% or greater radioactive methyl iodide removal when tested at at least 125°F and 95% RH and at

32.v

32.v

1.5 to 2.0 mg/m³ loading with tagged CH₃I.

4.5.2.3.7

After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the HEPA housing for the control room emergency air treatment system, the condition of Specification 4.5.2.3.6.c shall be demonstrated for the affected portion of the system.

4.5.2.3.8

After each replacement of a charcoal drawer or after any structural maintenance on the charcoal housing for the control room emergency air treatment system, the condition of Specification 4.5.2.3.6.b shall be demonstrated for the affected portion of the system.

Addressed in
Chapter 3.7

4.5.2.3.9 Except during cold or refueling shutdowns the automatic initiation of the control room emergency air treatment system shall be tested at intervals not to exceed one month to verify operability and proper orientation and flow shall be maintained through the system for at least 15 minutes. The test shall be performed prior to startup if the time since the last test exceeds one month.

Basis:

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a Safety Injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The annual systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action

and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.⁽¹⁾

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order and develop the minimum required pressure to meet accident conditions.⁽²⁾ The minimum discharge pressure values listed in Table 4.5-1 are based on an assumed degradation of the pump head-capacity (characteristic) curve adjusted to water temperature of 60°F as follows:

Containment Spray Pumps	5%*
Residual Heat Removal Pumps	5%*
Safety Injection Pumps	3%*

*Percentage is based on the head at the best efficiency point of flow.

The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required) and would result in increased wear over long periods of time.

Other systems that are also important to the emergency cooling function are the accumulators, the component cooling system, the service water system and the containment fan coolers. The accumulators are a passive safeguard. In accordance with the specifications, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance. The reactor coolant drain tank pumps operate intermittently during reactor operation, and thus are also monitored for satisfactory performance.

The air filtration portion of the containment air recirculation system is a passive safeguard which is isolated from the cooling air flow during normal reactor operation. Hence the charcoal should have a long useful lifetime. The filter frames that house the charcoal are stainless steel and should also last indefinitely. The pressure drop, filter efficiency, and valve operation test frequencies will assure that the system can operate to meet its design function under accident conditions. As the adsorbing charcoal is normally isolated, the test schedule, related to hours of operation as well as elapsed time, will assure that it does not degrade below the required adsorption

efficiency. The test conditions for charcoal sample adsorbing efficiency are those which might be encountered under an accident situation.⁽³⁾

The control room air treatment system is designed to filter the control room atmosphere (recirculation and intake air) during control room isolation conditions. HEPA filters are installed before the charcoal filters to remove particulate matter and prevent clogging of the iodine adsorbers. The charcoal filters reduce the airborne radioiodine in the control room. Bypass leakage must be at a minimum in order for these filters to perform their designed function. If the performances are as specified the calculated doses will be less than those analyzed.⁽⁴⁾

Retesting of the post accident charcoal system or the control room emergency air treatment system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.

Testing of the air filtration systems will be, to the extent it can, given the configuration of the systems, in accordance with ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems."

References:

- (1) UFSAR Section 6.3.5.2
- (2) UFSAR Figures 15.6-12 and 15.6-13
- (3) UFSAR Section 6.5.1.2.4
- (4) UFSAR Section 6.4.3.1

Addressed w/
Chapter 3.9

- c. The tests in Specification 4.6.1b will be performed prior to exceeding cold shutdown if the time since the last test exceeds 31 days.
- d. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment.

5.5.12
33.12

- e. At least once per 18 months during shutdown by:
 1. Inspecting the diesel in accordance with the manufacturer's recommendations for this class of standby service.
 2. Verifying the generator capability to reject a load of 295 KW without tripping.
 3. Simulating a loss of offsite power in conjunction with a safety injection test signal and:
 - (a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.
 - (b) Verifying the diesel starts from normal standby condition on the auto-start signal, energizes the automatically connected emergency loads with the following maximum breaker closure times after the initial starting signal for Trains A and B not being exceeded

Addressed w/
Chapter 3.9

	A	B
Diesel plus Safety Injection	20 sec	22 sec
Pump plus RHR Pump		
All Breakers	40 sec	42 sec

and operates for \geq five minutes while its generator is loaded with emergency loads.

- (c) Verifying that all diesel generator trips, except engine overspeed, low lube oil pressure, and overcrank, are automatically bypassed upon a safety injection actuation signal.

37.1

4.10 Radiological Environmental Monitoring

Applicability - Applies to routine testing of plant environs.

Objective - To establish a sampling and analysis program which will assure recognition of changes in radioactivity in the environs.

Specification

4.10.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.16-1. Acceptable locations are shown in the ODCM. Samples shall be analyzed pursuant to the requirements of Tables 3.16-1 and 4.10-1.

37.1

4.10.2 A land use census shall be conducted annually (between June 1 and October 1).

37.1

4.10.3 A summary of the results obtained as part of the required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report.

Basis

The environmental survey has been designed to utilize the knowledge about dilution in the atmosphere and in the lake which has been gained during the pre-operational and operational period of study.

The radiological monitoring program provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This

monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The detection capabilities required by Table 4.10-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The specified lower limits of detection for I-131 in water, milk, and other food products correspond to approximately one-quarter of the 10 CFR Part 50 Appendix I design objective dose-equivalent of 15 mrem/year for atmospheric releases and 10 mrem/year for liquid releases to the maximally exposed organ and individual.

Participation in an approved interlaboratory comparison program assures that the adequacy of environmental laboratory measurements is maintained on a continuing basis through independent cross-checking.

TABLE 4.10-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)^a

To be achieved on 98% of analyses

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m ³)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
gross beta	4 ^b	1 x 10 ⁻²			
³ H	2000 (1000 ^b)				
⁵⁴ Mn	15		130		
⁵⁹ Fe	30		260		
^{58,60} Co	15		130		
⁶⁵ Zn	30		260		
⁹⁵ Zr-Nb	15 ^c				
¹³¹ I	1	7 x 10 ⁻²		1	60
^{134,137} Cs	15(10 ^b), 18	1 x 10 ⁻²	130	15	60
¹⁴⁰ Ba-La	15 ^c			15 ^c	

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TABLE 4.10-1 (Continued)

TABLE NOTATION

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding its presence.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22 is the number of transformations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between sample collection and analysis

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculations.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small

37.2

TABLE 4.10-1 (Continued)

TABLE NOTATION

sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

- b - LLD for drinking water.
- c - Total for parent and daughter.

4.11

Refueling

Applicability

Applies to refueling and to fuel handling in the spent fuel pool.

Specification

4.11.1

Spent Fuel Pit Charcoal Adsorber System

4.11.1.1

Within 60 days prior to any operation of the spent fuel pool charcoal adsorber system as required by Section

38.c

5.5.10

3.11, the following conditions shall be demonstrated.

After the conditions have been demonstrated, the occurrence of painting, fire, or chemical release in any ventilation zone communicating with the spent fuel pool charcoal adsorber system shall require that the following conditions be redemonstrated, before fuel handling may continue, if operation of the spent fuel pool charcoal adsorber system is required per section 3.11

38.i

a. The total air flow rate from the charcoal adsorbers shall be at least 75% of that measured with a complete set of new absorbers.

38.ii

b. In-place Freon testing, under ambient conditions, shall show at least 99% removal.

c. The results of laboratory analysis on a carbon sample shall show 90% or greater radioactive methyl iodide removal when tested at least 150°F and 95% RH and at 1.5 to 2.0 mg/m³ loading with tagged CH₃I.

38.i

4.11

Addressed w/
Chapter 3.7

- d. Flow shall be maintained through the system using either the filter or bypass flow path for at least 15 minutes each month.

38.i 4.11.1.2 After each replacement of a charcoal filter drawer or after any structural maintenance on the charcoal housing for the spent fuel pit charcoal adsorber system, the condition of Specification 4.11.1.1.b shall be demonstrated for the affected portion of the system.

4.11.2 Residual Heat Removal and Coolant Circulation

4.11.2.1 When the reactor is in the refueling mode and fuel is in the reactor, at least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 4 hours.

4.11.2.2 When the water level above the top of reactor vessel flange is less than 23 feet, both RHR pumps shall be verified to be operable by performing the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a.

4.11.3 Water Level - Reactor Vessel

4.11.3.1 The water level in the reactor cavity shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods in containment.

Basis

The measurement of the air flow assures that air is being withdrawn from the spent fuel pit area and passed through the adsorbers. The flow is measured prior to employing the adsorbers to establish that

Addressed w/
Chapter 3.9

there has been no gross change in performance since the system was last used. The Freon test provides a measure of the amount of leakage from around the charcoal adsorbent.

The ability of charcoal to adsorb iodine can deteriorate as the charcoal ages and weathers. Testing the capacity of the charcoal to adsorb iodine assures that an acceptable removal efficiency under operating conditions would be obtained. The difference between the test requirement of a removal efficiency of 90% for methyl iodine and the percentage assumed in the evaluation of the fuel handling accident provides adequate safety margin for degradation of the filter after the tests.

Retesting of the spent fuel pit charcoal adsorber system in the event of painting, fire, or chemical release is required only if the system is operating and is providing filtration for the area in which the painting, fire, or chemical release occurs.

Testing of the air filtration systems will be tested, to the extent it can be given the configuration of the systems, in accordance with ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems".

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

Addressed w/ Chapter 3.9

Reference:

- (1) Letter from E. J. Nelson, Rochester Gas and Electric Corporation to Dr. Peter A. Morris, U.S. Atomic Energy Commission, dated February 3, 1971

39.i

4.12 Effluent Surveillance

Applicability

Applies to the periodic test and record requirements of the plant effluents.

Objective

To ascertain that radioactive liquid and gaseous releases from the plant are within allowable limits.

Specifications

4.12.1 Liquid Effluents

4.12.1.1 Concentration

4.12.1.1.a The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.12-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is limited to the values in Specification 3.9.1.1.a.

4.12.1.1.b Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.12-1. The results of the post-release analyses shall be used with the calculational methods in the ODCM to assure that the does commitments from liquids were limited to the values in Specification 3.9.1.2.a.

~~4.12.1.2 Dose, Liquid Waste Treatment~~

39.ii

39.ii

4.12.1.2.a Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

4.12.2 Gaseous Wastes

39.iii

4.12.2.1 Release Rate

4.12.2.1.a The effluent continuous monitors as listed in Table 3.5-6 having provisions for the automatic termination of gas decay tank, shutdown purge or mini-purge releases, shall be used to limit releases within the values established in Specification 3.9.2.1 when monitor setpoint values are exceeded.

4.12.2.1.b The dose rate due to radioactive materials, other than noble gases, in gaseous effluents shall be determined in accordance with the methods of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Table 4.12-2.

39.iv

4.12.2.2 Dose (10 CFR Part 50, Appendix I); Gaseous Waste Treatment.

4.12.2.2.a Cumulative dose contributions from gaseous effluents shall be determined in accordance with the ODCM at least once every 31 days.

4.12.3 Waste Gas Decay Tanks

The quantity of radioactive material contained in each waste gas decay tank shall be determined to be

39.v

4.12-3

(39.v)²

within the limit specified in 3.9.2.6.a at least once per 24 hours if the total primary coolant noble gas concentration exceeds 250 $\mu\text{Ci}/\text{gram}$ and primary coolant gas is being transferred to the gaseous radwaste treatment system.

Basis:

Sufficient tests will be made to be certain that radioactive materials are not released to the environment in quantities greater than allowable. Installed radiation monitoring equipment in the plant will be used in conjunction with laboratory analyses to maintain surveillance of normal effluents.

Sufficient records will be maintained to determine the concentration of radioactive materials in unrestricted areas. Isotopic analysis of representative samples will serve to verify the accuracy of routine samples by identification of significant energy peaks.

The quantity of radioactivity in each gas decay tank is determined when the noble gas concentration in the primary coolant system increases significantly enough to potentially contribute an appreciable quantity of noble gas activity to the gaseous radwaste system.

The required surveillance will be initiated at a primary noble gas concentration level which, if attained will still allow sufficient margin below the specified curie limit for a single gas decay tank.

Determination of tank curie content may be performed by sampling and/or calculation.

39.1

TABLE 4.12-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml) ^a
Batch Waste Release Tanks ^b	PR Each Batch	PR Each Batch	1. Principal Gamma Emitters and I-131	5 x 10 ⁻⁷ 1 x 10 ⁻⁶
			or	
			2. Gross beta-gamma*	5 x 10 ⁻⁷
	PR One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1 x 10 ⁻⁵
	PR Each Batch	M Composite ^c	H-3 Gross alpha	1 x 10 ⁻⁵ 1 x 10 ⁻⁷
	PR Each Batch	Q Composite ^c	Sr-89, Sr-90 Fe-55	5 x 10 ⁻⁸ 1 x 10 ⁻⁶
Continuous Release				
Retention Tank	Continuous	W Composite ^c	Principal Gamma Emitters and I-131	5 x 10 ⁻⁷ 1 x 10 ⁻⁶
Service Water (CV Fan Cooler and SFP HX lines)	Continuous	M or S** Grab	Gross beta-gamma	1 x 10 ⁻⁷

* If **gross** beta is performed for batch releases, then a weekly composite shall also be analyzed for Principal Gamma Emitters and I-131.

**Service water samples shall be taken and analyzed once per 12 hours if alarm setpoint is reached on continuous monitor.

39.1

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding its presence.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the lower limit of detection as defined above (as uCi per unit mass or volume)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22×10^6 is the number of transformations per minute per microcurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

39.1

The background count rate is calculated from the background counts that are determined to be within \pm one FWHM energy band about the energy of the gamma ray peak used for the quantitative analysis for this radionuclide.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. When circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Report.

- b. A batch release is the discharge of liquid wastes of a discrete volume.
- c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.
- d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, and Ce-141. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as less than the LLD and should not be reported as being present at the LLD level. The less than values should not be used in the required dose calculations. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.
- e. A continuous release is the discharge of liquid wastes of a non-discrete volume; e.g. from a volume of system that has an input flow during the continuous release.

3111

TABLE 4.12-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

<u>Gaseous Release Type</u>	<u>Sampling Frequency</u>	<u>Minimum Analysis Frequency</u>	<u>Type of Activity Analysis</u>	<u>Lower Limit of Detection (LLD) (uCi/ml)^a</u>
A. Gas Decay Tank	PR Each Tank Grab Sample	PR Each Tank	Principal Gamma Emitters ^e	1 x 10 ⁻⁴
B. Containment Purge	PR Each Purge ^{b,c} Grab Sample	PR Each Purge ^b	Principal Gamma Emitters ^e H-3	1 x 10 ⁻⁴ 1 x 10 ⁻⁶
C. Auxiliary Building Ventilation	M ^b Grab Sample	M ^b	Principal Gamma Emitters ^e H-3	1 x 10 ⁻⁴ 1 x 10 ⁻⁶
D. Air Ejector	M ^{b,f,h} Grab Sample	M ^b	Principal Gamma Emitters ^e , I-131 H-3 ^g	1 x 10 ⁻⁴ 1 x 10 ⁻⁶
E. All Release Types as listed in B and C above	Continuous ^d	W ^b Charcoal Sample	I-131 I-133	1 x 10 ⁻¹² 1 x 10 ⁻¹⁰
	Continuous ^d	W ^b Particulate Sample	Principal Gamma Emitters ^e (I-131, Others)	1 x 10 ⁻¹¹
	Continuous ^d	M Composite Particulate Sample	Gross alpha	1 x 10 ⁻¹¹
	Continuous ^d	Q Composite Particulate Sample	Sr-89, Sr-90	1 x 10 ⁻¹¹
F. All Release Types as listed in B, C and D above	Continuous ^d	Noble Gas Monitor	Beta or Gamma	1 x 10 ⁻⁶

4-12

TABLE 4.12-2 (Continued)

TABLE NOTATION

- a. The lower limit of detection (LLD) is defined in Table Notation a. of Table 4.12-1.
- b. Analyses shall also be performed when the monitor on the continuous sampler reaches its setpoint.
- c. Tritium grab samples shall be taken at least three times per week when the reactor cavity is flooded.
- d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with specifications 3.9.2.1.a, 3.9.2.2.a and 3.9.2.2.b.
- e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-85m, Xe-133, Xe-133m, and Xe-135 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.
- f. Air ejector samples are not required during cold or refueling shutdowns.
- g. Air ejector tritium sample not required if the secondary activity is less than 1×10^{-4} $\mu\text{Ci/gm}$.
- h. Air ejector iodine samples shall be taken and analyzed weekly if the secondary coolant activity exceeds 1×10^{-4} $\mu\text{Ci/gm}$.

39.iii

4.13 Radioactive Material Source Leakage Test

Applicability

Applies to the periodic test for leakage of radioactive material sources performed by the licensee or by other persons specifically authorized by the Commission or the state.

Objective

To ascertain that any leakage from radioactive material sources is sufficiently low.

Specifications

4.13.1 Sources which contain quantities of by-product material that exceed the quantities listed in 10 CFR 30.71 Schedule B and all other sources (including alpha emitters) containing greater than 0.1 microcuries shall be leak tested as follows:

- a. Except for sealed sources that are stored and not being used, and except for startup sources, each sealed source containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

40.6

40.0

- b. Sealed sources that are stored and not being used shall be tested for leakage prior to use or transfer to another user unless they have been tested within six months prior to the date of use or transfer. Sealed sources received from a transferor shall, in the absence of a certificate from the transferor indicating that a test has been performed within six months prior to the transfer, be tested prior to use.
- c. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

4.13.2 The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample. If the test reveals the presence of 0.005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations.

Basis

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those

quantities of radioactive by-product materials of interest to this specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39 (c) limits for plutonium.

4.14

Snubber Surveillance Requirements

4.14.1

Each snubber required by Specification 3.13 to be OPERABLE shall be demonstrated OPERABLE by the performance of the following inservice inspection program in addition to the requirements of Specification 4.2.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determine by Table 4.14-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.14-1.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or

41.i

4.14.1.c. (continued)

supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.14.1e. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirement shall be met.

41.i

41:2

TABLE 4.14-1

SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF UNACCEPTABLE SNUBBERS (Ref. Note 7)

Population or Category (Notes 1 and 2)	Column A	Column B	Column C
	Extend Interval (Notes 3 and 6)	Repeat Interval (Notes 4 and 6)	Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, this decision must be

TABLE 4.14-1 (continued)

documented before any inspection and shall be used as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: The provisions of Specification Section 4.0 are applicable for all inspection intervals up to and including 48 months.

Note 7: To determine the next surveillance interval, an unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is operable in its as-found condition by performance if a functional test and if it satisfies the acceptance criteria for functional testing.

41.i

4.14.1.d Functional Tests

At least once per 18 months during shutdown, a representative sample (at least 10% of the snubbers required by Specification 3.13) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.14.1e, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested. The representative sample selected for functional testing shall, as far as practical, include the various configurations, operating environments, range of sizes and capacities of snubbers.

In addition to the regular sample, snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test. Additionally, if a failed snubber has been repaired and reinstalled in another location, that failed snubber shall also be retested. These snubbers shall not be included in the regular sample.

If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

41.i

4.14.1.e.

Functional Test Acceptance Criteria

The snubber functional test shall verify that:

- 1) Activation (restraining action) is achieved within the specified range in both tension and compression;
- 2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;
- 3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and
- 4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement is verified.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis

An analysis shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this analysis

41.i

4.14.1.f. (continued)

shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers, irrespective of type, which may be subject to the same failure mode. For the specific case of a snubber selected for functional testing which either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested or evaluated in a manner to ensure their operability. Any testing performed as part of this requirement shall be independent of the requirements stated in Specification 4.14.1d for snubbers not meeting the functional test acceptance criteria.

For any snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

41.i ~

4.14.1.g Snubber Seal Service Life Monitoring

The seal service life of hydraulic snubbers shall be monitored and seals replaced as required to ensure that the service life is not exceeded between surveillance inspections during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance with Technical Specification 6.10.2

(4.13)

Basis

Snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The visual inspection frequency is based on the number of unacceptable snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories. A snubber is considered unacceptable if it fails the acceptance criteria delineated by Specification 4.14.1.c. The visual inspection interval is based upon the previous inspection interval and may be as long as two fuel cycles, not to exceed 48 months, depending on the number of unacceptable snubbers found during the previous visual inspection.

Basis (continued)

Unacceptable snubbers shall be evaluated to determine if they are inoperable. For inoperable snubbers the applicable action requirements shall be met. When a snubber is found inoperable, an engineering evaluation of the supported component is performed in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. This evaluation is in addition to the determination of the snubber mode of failure. The engineering evaluation shall determine whether or not the snubber failure has imparted a significant effect on or caused degradation of the supported component or system, to ensure they remain capable of meeting the designed service.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the snubber rejected or are those which are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. To determine the next surveillance interval, an unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is operable in its as-found condition by performance of a functional test and if it satisfies the acceptance criteria for functional testing.

Basis (continued)

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at less than or equal to 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs. The service life of a snubber is evaluated via manufacturer input and engineering information through consideration of the snubber service conditions and functional design requirements. The only snubber components with service lives not expected to exceed plant life are seals and o-rings fabricated from certain seal materials. Therefore, a seal replacement program is required to monitor snubber seal and o-ring service life to assure snubber operability is not degraded due to exceeding component service life.

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~~Amendment No. 49 4.15-1]~~

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager, Ginna Station shall be responsible for overall on-site Ginna Station operation and shall delegate in writing the succession to this responsibility during his absence.

49.iii
5.1.1

49.ii

49.ii — Add TS 5.1.2

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organization

5.2.1

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all Plant management positions. Those relationships shall be documented and updated, as appropriate, in the form of organization charts. These organization charts ~~will~~

including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these technical specifications shall

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be documented in the UFSAR and updated in accordance with 10 CFR 50.71.

50.i

b. ~~The Senior Vice President, Customer Operations*~~

5.2.1.c

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shall have corporate responsibility for overall Plant nuclear safety, and shall take any measures needed to assure acceptable performance of the staff in operating, maintaining, and providing technical support in the Plant so that continued nuclear safety is assured.

A corporate vice president

* An alternate title may be designated for this position in accordance with 10 CFR 50.54(a)(3). All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

50.ii

Shall report to the Corporate Vice President specified in 5.2.1.c,

5.2.1.b

c. The Plant Manager, Ginna Station shall have responsibility for overall unit operation and shall have control over those resources necessary for safe operation and maintenance of the Plant.

Individuals who train the operating staff, carry out

d. The persons responsible for the training, health physics, ^{or} and quality assurance functions may report to ^{the} an appropriate manager onsite, ^{however, these individuals} but shall have

50.iii

direct access to responsible corporate management at a level where action appropriate to the mitigation of training, health physics and quality assurance concerns can be accomplished.

Sufficient organizational freedom to ensure their independence from operating procedures.

6.2.2 Facility Staff

The Facility organization shall include the following:

5.2.2.a

a. An auxiliary operator shall be assigned to the shift crew with fuel in the reactor. An additional auxiliary operator shall be assigned to the shift crew above Cold Shutdown.

50.iv

b. At least one licensed operator shall be present in the control room when fuel is in the reactor. In addition, above Cold shutdown, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

5.2.2.b

c. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.2.a and 6.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore shift crew composition to within the minimum requirement.

5.2.2.c

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d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

5.2.2.d

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e. Adequate shift coverage shall be maintained without routine heavy use of overtime. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions including senior reactor operators, reactor operators, health physicists*, auxiliary operators, and key maintenance personnel. Changes to the guidelines for the administrative procedures shall be submitted to the NRC for review.

5.2.2.f

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be assigned to the shift crew above Cold Shutdown.

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5.2.2.e The operations manager or operations middle manager shall hold a SRO license.

50.ii

* An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

6.3 STATION STAFF QUALIFICATIONS

6.3.1 Each member of the facility shall meet or exceed the
 5.3.1 minimum qualifications of ANSI Standard N18.1-1971,
 "Selection and Training of Nuclear Power Plant Personnel",
 as supplemented by Regulatory Guide 1.8, September
 1975, for comparable positions, except for the Shift
 5.2.2: 51.i - Technical Advisor.¹

References

1 Ltr. J. Maier (RG&E) to D. Crutchfield (NRC), dated
 December 30, 1980.

51.i

52.i

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Division Training Manager* and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55.

6.4.2 The training program shall meet or exceed NFPA No. 27, 1975 Section 40, except that (1) training for salvage operations need not be provided and (2) the Fire Brigade training sessions shall be held at least quarterly. Drills are considered to be training sessions.

* An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

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~~Amendment No. 1, 78, 77, 77,
78, 48, 49, 58~~ 6.5-1e

6.6 ~~(Deleted)~~

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~~Amendment No. 4, 77, 78, 58 — 6.6-1~~

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the following activities:

- 5.4.1 a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.
- b. Fire Protection Program implementation.

5.5.4 (5.6.v) c. The radiological environmental monitoring program.

5.5.1 (5.6.i) d. Offsite Dose Calculation Manual implementation.

(5.6.ii) e. Process Control Program implementation.

(5.6.iii) — Add TS 5.4.1.b

(5.6.iv) — Add TS 5.4.1.2

6.9

Reporting Requirements

5.6

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the USNRC, Region 1, unless otherwise noted.

57.0

6.9.1

Routine Reports

6.9.1.1

Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests performed and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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~~Startup reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption of commercial power operation, whichever is earliest. If the Startup Report does not cover both events (i.e., completion of startup test program, and resumption of commercial power operation), supplementary reports shall be submitted at least every three months until both events have been completed.~~

5.6.4

6.9.1.2

Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted ~~in accordance with 10 CFR 50.4 on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555~~ by the fifteenth of each month following the calendar month covered by the report.

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~~The monthly report shall include a narrative summary of operating experience describing the operation of the facility, including major safety related maintenance for the monthly period, except that safety related maintenance performed during the refueling outage may be reported in the monthly report for the month following the end of the outage rather than each month during the outage.~~

5.6.5

6.9.1.3

Annual Radiological Environmental Operating Report

A radiological environmental operating report covering the operation of the unit during the previous calendar year shall be submitted prior to May ²/₁₅ of each year.

5.3.2

The annual radiological environmental report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with background (control) samples and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses as required.

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5.6.2

The annual radiological environmental operating report shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report. In addition, the annual report

57.2v

shall include a discussion which identifies the circumstances which prevent any required detection limits for environmental sample analyses from being met, and a discussion of all deviations from the sample schedule of Table 3.16-1. The report shall also include the following: a summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances

57.iv

and directions from the reactor, and the results of the participation in an interlaboratory comparison program.

6.9.1.4 Radioactive Effluent Release Report

5.6.3

57.v

Routine radioactive effluent release reports covering the operation of the unit during the previous twelve months of operation shall be submitted by May 1 of each year. This report shall include a summary, on a quarterly basis, of the quantities of radioactive liquid and gaseous effluents and solid waste released as outlined in Regulatory Guide 1.21, Revision 1.

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This report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each of the previous four calendar quarters as outlined in Regulatory Guide 1.21, Revision 1. In addition, the site boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM. This same report shall include an annual summary of hourly meteorological data collected over the previous calendar year. Alternatively, the licensee has the option of retaining this summary on site in a file that shall be provided to the NRC upon request.

Also, the report shall include any nearby location(s) identified by the land use census which

yield a calculated dose or dose commitment greater than those forming the basis of Specifications 4.12.2.2 or 3.16.1. The report shall also contain a discussion which identifies the causes of the unavailability of milk or leafy vegetable samples and identifies locations for obtaining replacement samples in accordance with Specification 3.16.1.4.

The radioactive effluent release report shall include a discussion which identifies the circumstances which prevent any required detection limits for effluent sample analyses from being met.

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The radioactive effluent release reports shall include any changes made during the reporting period to the ODCM as specified in Section 6.15, and to the Process Control Program as specified in Section 6.16. The radioactive effluent release reports shall also include a discussion of any major changes to radioactive waste treatment systems in accordance with Specification 6.17.2.1.

6.9.1.5 Pressurizer Relief and Safety Valve Challenges

5.6.4 Challenges to the pressurizer power operated relief valves or safety valves shall be reported no less frequently than on an annual basis.

57.vi

~~6.9.2 Unique Reporting Requirements~~

6.9.2.1 ~~Annually: Results of required leak test performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.~~

57.vii

6.9.2.2 Annually: A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. (NOTE: This tabulation supplements the requirements of Section 20.407 of 10CFR Part 20)

6.9.2.3 ~~(Deleted)~~

5.6.1

6.9.2.4 Reactor Overpressure Protection System Operation

In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission within thirty days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any other corrective action necessary to prevent recurrence.

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57.ix - Add TS 5.6.5

57.x - Add TS 5.6.6

57. iv

TABLE 6.9-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility R. E. Ginna Nuclear Power Plant Docket No. 50-244

Location of Facility Wayne County, New York Reporting Period _____

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection ^a (LLD)	All Indicator Locations Mean (1) ^b Range	Locations with Highest Name Distance and Direction	Annual Mean Mean(1) ^b Range	Control Location Mean(1) ^b Range
---	--	--	---	--	--	---

^a Nominal Lower Limit of Detection (LLD) as defined in Table Notation a. of Table 4.12-1.

^b Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses (1).



(57.1v)

TABLE 6.9-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m ³)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Broad Leaf Vegetables (pCi/Kg, wet)
H-3	2 x 10 ⁴				
Mn-54	1000		3 x 10 ⁴		
Fe-59	400		1 x 10 ⁴		
Co-58	1000		3 x 10 ⁴		
Co-60	300		1 x 10 ⁴		
Zn-65	300		2 x 10 ⁴		
Zr-Nb-95	400 (a)				1
I-131	2	0.9		3	1 x 10 ²
Cs-134	30	10	1 x 10 ³	60	1 x 10 ³
Cs-137	50	20	2 x 10 ³	70	2 x 10 ³
Ba-La-140	200 (a)			300	

(a) Total for parent and daughter

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~~Amendment No. 1, 20, 77, 58 6.10-1~~

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~~Amendment No. 58~~

~~6.11-E~~

6.12 ~~(Deleted)~~

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6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20:

5.7.1

a. Each High Radiation Area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit* (RWP). Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

(1) A radiation monitoring device which continuously indicates the radiation dose rate in the area.

(2) A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

* Radiation Protection** personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, providing they are following plant radiation protection procedures for entry into high radiation areas.

** An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

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5.7.1

(3) A Qualified health physicist* (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who will perform periodic radiation surveillance at the frequency specified in the HPWP. The surveillance frequency will be established by a plant health physicist*.

5.7.2
5.7.3

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1 a. above, and in addition, locked doors shall be provided to prevent unauthorized entry into these areas and the keys to unlock these locked doors shall be maintained under the administrative control of the Shift Supervisor on duty.

* An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

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~~Amendment No. 4~~



5.5.1

6.15 Offsite Dose Calculation Manual (ODCM)

6.15.1 Any changes to the ODCM shall be made by the following method:

6.15.1.a Licensee initiated changes shall be submitted to the Commission with the Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

5.5.1.c
5.5.1.a

- (i) sufficiently detailed information to support the rationale for the change.
- (ii) a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and
- (iii) documentation of the fact that the change has been reviewed and found acceptable by the onsite review function.

5.5.1

6.15.1.b Licensee initiated changes shall become effective after review and acceptance by the onsite review function on

5.5.1.b

43.i a date specified by the licensee.



6.16 Process Control Program (PCP)

6.16.1 Any changes to the PCP shall be made by the following method:

6.16.1.a Licensee initiated changes shall be submitted to the Commission with the Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

- (i) sufficiently detailed information to support the rationale for the change;
- (ii) a determination that the change will not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
- (iii) documentation of the fact that the change has been reviewed and found acceptable by the onsite review function.

6.16.1.b Licensee initiated changes shall become effective after review and acceptance by the onsite review function on a date specified by the licensee.

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6.17 Major Changes to Radioactive Waste Treatment Systems
(Liquid, Gaseous and Solid)

FUNCTION

- 6.17.1 The radioactive waste treatment systems (liquid, gaseous and solid) are those systems defined in Technical Specification 5.5.
- 6.17.2 Major changes to the radioactive waste systems (liquid and gaseous) shall be reported by the following method. For the purpose of this specification, "major changes" is defined in Specification 6.17.3 below.
 - 6.17.2.1 The Commission shall be informed of all major changes by the inclusion of a suitable discussion or by reference to a suitable discussion of each change in the Radioactive Effluent Release Report for the period in which the changes were made. The discussion of each change shall contain:
 - a) a summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);
 - b) sufficient detailed information to support the reason for the change;
 - c) a detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

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SECRET

TO: SAC, NEW YORK (100-100000) FROM: SAC, PHOENIX (100-100000) (P)

RE: [REDACTED] (C) (U) (S) (P)

DATE: 10/15/68

BY: [REDACTED]

CLASSIFICATION: [REDACTED]

EXEMPTION: [REDACTED]

REASON: [REDACTED]

DATE OF REVIEW: [REDACTED]

BY: [REDACTED]

SECRET

BT

- d) an evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents from those previously predicted;
- e) an evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population from those previously estimated;
- f) documentation of the fact that the change was reviewed and found acceptable by the PORC.

6.17.3

"Major Changes" to radioactive waste systems (liquid, gaseous and solid) shall include the following:

- a) Major changes in process equipment, components, and structures from those in use (e.g., deletion of evaporators and installation of demineralizers);
- b) Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly alter the characteristics and/or quantities of effluents released;
- c) Changes in system design which may invalidate the accident analysis (e.g., changes in tank capacity that would alter the curies released).

(65.i)

~~6.17.2~~

