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WATTS BAR NUCLEAR PLANT
PROPOSED TECHNICAL SPECIFICATIONS

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5.0 ADMINISTRATIVE CONTROLS

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LIST OF ACRONYMS

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling Water System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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1.1 DEFINITIONS

------------------------------------NOTE------------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases
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Term	Definition
ACTIONS	ACTIONS shall be that part of a Technical Specification which 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 input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
ANALOG CHANNEL OPERATIONAL TEST	An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock, and trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and trip setpoints such that the setpoints are within the required range and accuracy.
AXIAL FLUX DIFFERENCE (AFD)	AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a 2 section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor, alarm, interlock and trip functions. It may be performed by any series of sequential, overlapping calibrations, or total channel steps such that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and status with other indications and status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 DEFINITIONS (continued)

CORE ALTERATION	A CORE ALTERATION shall be the addition, removal, relocation, or movement of any fuel, sources, reactivity control or other reactor vessel components that would affect reactivity 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 CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 5.10.8. Plant operation within these core operating limits is addressed in individual specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) 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 thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites", [or those listed in Table E-7 of NRC Regulatory Guide 1.109, [Rev. 1, October 1988].
$\bar{\bar{E}}$-AVERAGE DISINTEGRATION ENERGY	$\bar{\bar{E}}$ 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 per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than [15] minutes, making up at least 95 percent of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

(continued)

1.1 DEFINITIONS (continued)

LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE, except reactor coolant pump seal water injection or leakoff, such as pump seal or valve packing leaks, that is captured and conducted to a collection system, or 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 LEAKAGE through a steam generator to the secondary system. b. Pressure Boundary LEAKAGE 1. Leakage (except steam generator tube LEAKAGE) through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall. c. Unidentified LEAKAGE 1. All LEAKAGE which is not Identified LEAKAGE or reactor coolant pump seal water injection or leakoff.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, train, component or device shall be OPERABLE when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified function(s) are also capable of performing their related support function(s).

(continued)

1.1 DEFINITIONS (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure nuclear characteristics important to validate the Safety Analysis; and which are: a. Described in Chapter [14.0, Initial Test Program] of the FSAR, b. Authorized under the provisions of 10 CFR 50.59, or c. Otherwise approved by the Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PRESSURE AND TEMPERATURE LIMITS REPORT is the unit-specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.10.9. Plant operation within these operating limits is addressed in LCO 3.4.3 Reactor Coolant System Pressure/Temperature Limits.
QUADRANT POWER TILT RATIO (QPTR)	QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.
RATED THERMAL POWER (RTP)	RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of [3411] MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

(continued)

1.1 DEFINITIONS (continued)

SHUTDOWN MARGIN	SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All control rods and shutdown rods are fully inserted, except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn. b. In Mode 1 and 2, the fuel and moderator temperature are changed to the nominal zero power level. c. In addition, with a control rod not capable of being fully inserted, the reactivity worth of this control rod must be accounted for in the determination of SHUTDOWN MARGIN.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.
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 specified surveillance frequency such 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	TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and trip functions.

Table 1.1-1

(Page 1 of 1)

Modes

MODE	TITLE	REACTIVITY CONDITION, Keff	%RATED THERMAL POWER**	AVERAGE REACTOR COOLANT TEMPERATURE °F
1	Power Operation	$\geq$ 0.99	> 5	$\geq$ 350
2	Startup	$\geq$ 0.99	$\leq$ 5	$\geq$ 350
3	Hot Standby	< 0.99	NA	$\geq$ 350
4	Hot Shutdown	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown	< 0.99	NA	$\leq$ 200
6	Refueling *	$\leq$ 0.95	NA	[$\leq$ 140]

* Fuel in the reactor vessel with one or more reactor vessel head closure bolts less than fully tensioned or with the head removed.

** Excluding decay heat.

1.2 LOGICAL CONNECTORS

PURPOSE

Logical connectors are used in technical specifications to discriminate between, and yet connect, discrete, Conditions, Required Actions and Completion Times, Surveillances, and Frequencies. The only logical connectors which appear in technical specifications are AND and OR. The physical arrangement of these connectors constitute logical conventions with specific meaning.

The purpose of this section is to explain the intended meaning of logical connectors and provide specific examples.

EXAMPLE 1.2.1

This example, as illustrated below, demonstrates that for Condition A, both Required Actions must be completed. In this case the logical connector AND is left justified and shows that both the Required Actions A.1 and A.2 shall be performed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.	A.1 Restore... AND A.2 Be in...	

1.2 LOGICAL CONNECTORS (continued)

EXAMPLE 1.2.2

This example, as illustrated below is a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3.1 are alternate choices. Either may be chosen. If A.3.1 is chosen, an additional requirement, indicated by the indented logical connector <u>AND</u>, is imposed. This additional requirement is met by choosing A.3.2.1 or A.3.2.2. The indented position of the logical connector <u>OR</u> indicates that A.3.2.1 and A.3.2.2 are alternate and equal choices, one (but not both) of which shall be performed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.	A.1 Restore ... <u>OR</u> A.2 Align ... <u>OR</u> A.3.1 Verify ... <u>AND</u> A.3.2.1 Reduce ... <u>OR</u> A.3.2.2 Reduce ...	

1.3 COMPLETION TIMES

PURPOSE

When a Limiting Condition for Operation (LCO) or Safety Limit is not met, Technical Specifications identify Required Actions which must be completed within specified times. An understanding of the correct meaning of Completion Times is necessary for compliance with the requirements of the Technical Specifications.

The purpose of this section is to discuss the use of these Completion Times, and explain, by example, the proper meaning.

COMPLETION TIME

The Completion Time is the amount of time allowed to complete a Required Action and is referenced to the time it is discovered that an LCO is not satisfied, unless otherwise specified. Required Actions must be completed prior to the expiration of the specified Completion Time. If the Required Action is not completed within the specified Completion Time, then the Required Action is considered incomplete.

EXAMPLE 1.3.1

When a Required Action is composed of two or more requirements, they are linked by logical connectors (i.e., <u>AND</u> or <u>OR</u>) and each requirement has its own separate Completion Time.

(1) If the requirement is to be in MODE 3 in 6 hours <u>AND</u> in MODE 5 in 36 hours, a total of 6 hours is allowed to reach MODE 3 and a total of 36 hours (<u>not</u> 42 hours) is allowed to reach MODE 5 from the time that the LCO was entered. If MODE 3 is reached in 3 hours, the time allowed to reach MODE 5 is the next 33 hours because the total time to reach MODE 5 is not reduced from the allowable limit of 36 hours.

(2) If the requirement is to be in MODE 3 in 6 hours <u>AND</u> MODE 5 in 36 hours and the LCO is entered in MODE 3 or MODE 4, the time allowed to reach MODE 5 is the next 36 hours because the total time to reach MODE 5 is not reduced from the allowable limit of 36 hours.

(3) If the requirement is to restore a process variable to within its limits in 1 hour <u>OR</u> be MODE 3 in 6 hours, 1 hour is allowed to restore the parameter or 6 hours (<u>NOT</u> 7 hours) is allowed to reach MODE 3 from the time that the ACTIONS Condition was entered.

(continued)

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump INOPERABLE	A.1 Restore pump to OPERABLE status	7 days
B. Two pumps INOPERABLE	B.1 Restore one pump to OPERABLE status	72 hours

EXAMPLE 1.3.2.1 When the ACTIONS Condition changes, a new Completion time is established for the new ACTIONS Condition, but the Completion Time for the original Condition still exists. In example 1.3.2, when the Condition changes from a single inoperable pump (Condition A) to two inoperable pumps (We are now in Condition A and Condition B at the same time), the Completion Time for one inoperable pump continues and a separate Completion Time is established for the two pump inoperable Condition. Thus, if the Required Action for the second Condition (B.1) is completed, and the Completion Time for the original Condition (A.1) has not expired, the unit is still in the original Condition, with a Completion Time measured from the original time of entry into the LCO. This does not reset the time of entry into Condition A itself, and Required Action A.1 must still be met within its original time frame.

(continued)

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.2.1 (continued)

To illustrate,

At time t-0, pump #1 inoperable, enter Condition A Start clock, and set trip point t_{tp} = 7 days

If nothing changes, clock will trip at 7 days

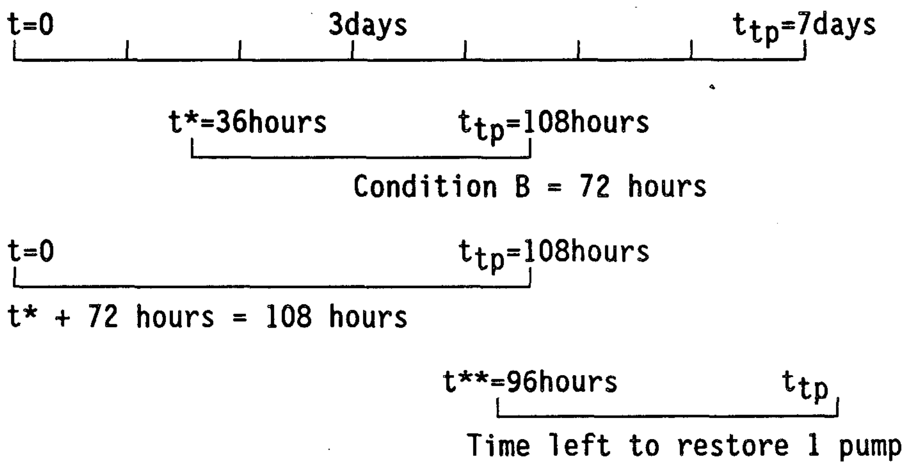
At time t* = 36 hours, pump #2 inoperable, enter Condition B.

At this point we are Condition A <u>and</u> Condition B and we must take all the Required Actions of both Conditions and we must meet the most restrictive Completion Times of both Conditions.

Do not reset start clock, <u>reset</u> trip point t_{tp} = t* + 72 hours (36 + 72) = 108 hours

At time t** = 96 hours (4 days), 1 pump restored, Exit Condition B, however, still in Condition A.

Do not reset start clock, There are only <u>3</u> days left of the 7 day Completion Time to restore the one pump <u>reset</u> trip point t_{tp} = 7 days - 96 hours



(continued)

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.2.2 If the Completion Time for the original Condition (one pump inoperable) expires while still in the new Condition (two pumps inoperable), then the Required Action associated with the original Condition is incomplete.

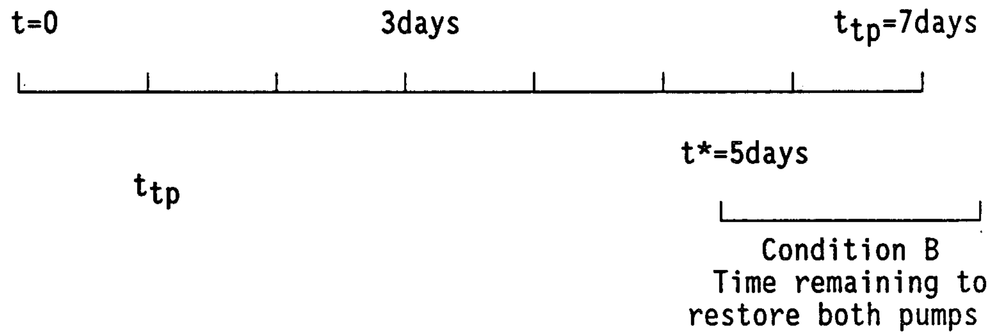
To illustrate,

At time t=0, pump #1 inoperable, enter Condition A.

Start clock, set trip point t_{tp} = 7 days

At time t*=120 hours (5 days), pump #2 inoperable, enter Condition B.

Do not reset start clock, reset trip point t_{tp}=t* + 72 hours = 192 hours or 8 days (Note that the 72 hours allowed for Condition B cannot be used because it exceeds the 7 days allowed for Condition A. Therefore, note that only 48 hours, not 72 hours, remain to restore both inoperable pumps).



If at time t**=168 hours (7 days), 1 pump is restored, there is not time left to restore second pump since t**=7days. So, the ACTION is incomplete.

(continued)

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.3
(SIMPLIFIED)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of the required offsite circuits inoperable	LCO 3.8.1 - A	72 hours
B. One of the required diesel generators inoperable	LCO 3.8.1 - B	72 hours
C. One of the required offsite circuits inoperable AND One of the required diesel generators inoperable	LCO 3.8.1 - C	12 hours
D. Two of the required diesel generators inoperable	LCO 3.8.1 - D	2 hours
E. Two of the required offsite circuits inoperable	LCO 3.8.1- E	24 hours
F. Required Actions note met within the required Completion Times	Be in MODE 3 AND Be in Mode 5	6 hours 36 hours

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.3 (continued)

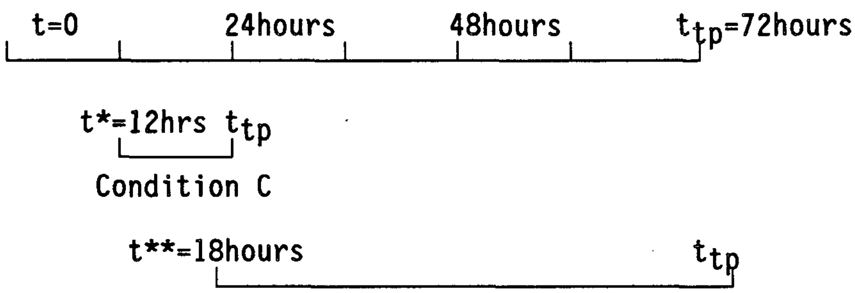
One of the required offsite circuits becomes inoperable, Condition A exists with a Completion Time for one inoperable offsite circuit. If, while one of the required offsite circuits is still inoperable, one of the required diesel generators also becomes inoperable, Condition C exists (two inoperable components of different types), thus establishing a new Completion Time. If the Required Action for the first Condition is completed (offsite circuit restored), Condition B (one inoperable diesel generator) exists. The Completion Time for Condition B is measured from the original time of entry into Condition A and must still be met within the Completion Time specified for Condition B.

To illustrate,

At time t=0, offsite circuit #1 inoperable,
Enter Condition A,
Start clock,
Set trip point at t_{tp} = 72 hours,

At time t*=12 hours, diesel generator #1 inoperable,
Enter Condition C,
Do <u>not</u> reset start clock,
Reset trip point $t_{tp} = t^* + 12$; $t_{tp} \leq 72$.

At time t**= 18 hours, restore offsite circuit #1,
Enter Condition B,
Do <u>not</u> reset start clock,
Trip point t_{tp} = 72 hours.



Note that there are only 54 hours left to be in Condition B [72 - 18 = 54 hours].

(continued)

1.3 COMPLETION TIMES (continued)

EXAMPLE 1.3.4

An exception to the Completion Time being reference to the time an LCO is first not satisfied is noted by the user of "thereafter" following the Completion Time. In this instance, the Completion Time might read "24 hours AND once per 12 hours thereafter." The 24 hour Completion Time begins at the time the LCO was first not satisfied; the "and once per 12 hours thereafter" begins after the 24 hour Completion Time was satisfied. This requires the initiation of periodic performance of the Required Action after the successful completion of the initial performance.

EXAMPLE 1.3.5

Occasionally "Immediately" is used as a Completion Time. "Immediately" is generally used when a Condition cannot be permitted to continue, and little time can be tolerated to take the Required Action. When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and before an attempt is made to correct the Condition. In some of these cases a Required Action must be initiated immediately but cannot be completed for an extended period of time. Then a two part construction is used:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
	Initiate ...	Immediately
	AND	
	Continue ...	Until Complete or x hours

1.4 FREQUENCY

PURPOSE

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met to support meeting the associated Limiting Condition for Operation (LCO). An understanding of the correct application of this Frequency is necessary for compliance with the Surveillance Requirements.

This purpose of this section is to discuss the proper use and application of the Frequency.

FREQUENCY

In examples provided and discussed below, the applicability of the LCO is given as MODES 1, 2, 3, and [4].

EXAMPLE SR 1.4.1

Example SR 1.4.1 contains the type of the SR most often encountered throughout the Technical Specifications. It specifies an interval (12 hours) during which the associated Surveillance Requirements 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. The measurement of this interval continues at all times, even when the Surveillance Requirement is not required by SR 3.0.1. In cases where the interval specified by SR 3.0.2 is exceeded while in a MODE or other specified condition for which the performance of the Surveillance Requirement is required, SR 3.0.3 becomes applicable. In cases where the interval as specified by SR 3.0.2 is exceeded while not in a MODE or other specified condition for which performance of the Surveillance Requirement is required, the Surveillance Requirement must be performed prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4 and an invalid MODE change per LCO 3.0.4.

SR 1.4.1 Perform CHANNEL CHECK	12 hours

(continued)

1.4 FREQUENCY

PURPOSE

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met to support meeting the associated Limiting Condition for Operation (LCO). An understanding of the correct application of this Frequency is necessary for compliance with the Surveillance Requirements.

This purpose of this section is to discuss the proper use and application of the Frequency.

FREQUENCY

In examples provided and discussed below, the applicability of the LCO is given as MODES 1, 2, 3, and [4].

EXAMPLE SR 1.4.1

Example SR 1.4.1 contains the type of the SR most often encountered throughout the Technical Specifications. It specifies an interval (12 hours) during which the associated Surveillance Requirements 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. The measurement of this interval continues at all times, even when the Surveillance Requirement is not required by SR 3.0.1. In cases where the interval specified by SR 3.0.2 is exceeded while in a MODE or other specified condition for which the performance of the Surveillance Requirement is required, SR 3.0.3 becomes applicable. In cases where the interval as specified by SR 3.0.2 is exceeded while not in a MODE or other specified condition for which performance of the Surveillance Requirement is required, the Surveillance Requirement must be performed prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4 and an invalid MODE change per LCO 3.0.4.

SR 1.4.1 Perform CHANNEL CHECK	12 hours

(continued)

1.4 FREQUENCY (continued)

FREQUENCY (continued)

Sometimes special circumstances dictate when a Surveillance is to be met. These conditions may apply to the Surveillance, or to the Frequency, or to both. They are the "otherwise stated" conditions allowed by SR 3.0.1. Furthermore, these conditions may be stated as clarifying notes or as part of the Surveillance Requirement itself. The following examples discuss these special circumstances.

EXAMPLE SR 1.4.2

Example SR 1.4.2 is a Surveillance with compound Frequency requirements which include a conditional event Frequency (Within 15 minutes prior to...) followed by a Frequency as described in SR 1.4.1 (12 hours). The logical connector "AND" requires both Frequencies to be met. If no other guidance is given, "prior to" means "within the specified Frequency prior to", and only requires the Surveillance be performed once during this period. Sufficient guidance is provided with this conditional event Frequency to determine the time interval within which the Surveillance must be performed. Since the conditional event Frequency in this example is performed only once ("prior to" the event) the Frequency extension allowance of SR 3.0.2 does not apply to the 15 minutes. Should the conditional event (initial control bank withdrawal...) not occur prior to the Frequency (15 minutes) elapsing, the Surveillance must be performed again such that the Surveillance is performed within 15 minutes of the conditional event. The Sample Frequency of 12 hours is then applied thereafter as described in Examples SR 1.4.1.

SR 1.4.2	Verify each shutdown bank within limit.	within 15 minutes prior to initial control bank withdrawal during an approach to criticality AND 12 hours

(continued)

1.4 FREQUENCY (continued)

EXAMPLE SR 1.4.4

Example SR 1.4.4 requires that the Surveillance be performed only above 20% RTP. The phrase "Only required..." means this Surveillance may be performed in any MODE or other specified condition where unit status would allow successful completion, but is not required to be performed unless greater than or equal to 20% RTP.

The interval measurement for the frequency of this Surveillance continues at all times, as described in Example 1.4.1. However, if this Surveillance did not meet SR 3.0.2 while operation continued at less than 20% RTP, it would not constitute a failure to meet the LCO. The Surveillance is not required below 20% RTP, even though the LCO, per its Applicability, may be required to be met. Prior to reaching 20% RTP, if the Surveillance were not performed within the interval as allowed by SR 3.0.2, it must still be performed prior to reaching 20% RTP. If it is not performed prior to exceeding 20% RTP, the provisions of SR 3.0.3 would apply.

SR 1.4.4	-----------NOTE-------------- Only required with THERMAL POWER > 20% RTP ----------------------------- Verify linear heat rate within limits.	7 days

1.5 LEGAL CONSIDERATIONS

INTRODUCTION

The Atomic Energy Act of 1954 requires that technical specifications be a part of operating licenses. As such, they are enforceable under federal statute as well as under Title 10 Code of Federal Regulations (CFR). When an applicant receives a license from the Nuclear Regulatory Commission to operate a commercial nuclear power plant, the technical specifications are included as Appendix A to the license. Consequently, whenever a change is made to a plant's technical specifications, it requires an amendment to the operating license.

There are, however, certain sections and additional items included with the technical specifications that are not legally a part of the technical specifications or the operating license. This section identifies the legal parts (i.e., the additional items that require a license amendment to make a change) of technical specifications and those additional parts that do not require a license amendment.

LEGAL PARTS

10 CFR 50.36 delineates those items which are to be included in the technical specifications. Items to be included for nuclear power plants are:

- Safety Limits
- Limiting Safety System Settings
- Limiting Conditions for Operation
- Surveillance Requirements
- Design Features
- Administrative Controls

In addition, the Use and Application Division which is comprised of Definitions, Logical Connectors, Completion Times and Legal Considerations, is also a legal part of the technical specifications.

Since the technical specifications are normally issued as Appendix A to the operating license, any change to the legal parts of the technical specifications constitutes a license amendment. As such, the requirements contained in 10 CFR 50.90, 50.91 and 50.92 apply.

(continued)

1.5 LEGAL CONSIDERATIONS (continued)

FRONT MATTER

Front matter is all the material in the front of the technical specifications used to identify and locate specific information. It may include:

- o Preface
- o Title Page
- o Table of Contents
- o List of Tables
- o List of Figures
- o List of Effective Pages

None of this material is required by 10 CFR 50.36 and it does not include any requirements on the safe operation of the plant. Therefore, this section is not a legal part of the technical specifications or the operating license.

CROSS REFERENCES

Cross-References are included in the body of the technical specifications to assist the user in determining applicable requirements for a common system or component. This section of technical specifications is not mandated by 10 CFR, and cross-references are included in the technical specifications at the discretion of the licensee. As such, they are not a legal part of the technical specifications or the operating license.

BASES

10 CFR 50.36 includes the following statement, "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall be included in the application, but shall not become part of the technical specifications". Therefore, the bases sections are not a legal part of the technical specifications.

2.0 SAFETY LIMITS

2.1 SAFETY LIMITS

2.1.1 The combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the Safety Limits specified in Figure 2.1.1-1, in MODES 1 and 2.

2.1.2 The RCS pressure shall be $\leq$ [2735] psig in MODES 1, 2, 3, 4, and 5.

2.2 SAFETY LIMIT VIOLATION

2.2.1 In MODES 1 or 2, restore compliance with all Safety Limits and be in MODE 3 within 1 hour.

2.2.2 In MODES 3, 4, or 5, restore RCS pressure Safety Limit compliance within 5 minutes.

2.2.3 Within 1 hour, notify the NRC Operations Center in accordance with 10 CFR 50.72.

2.2.4 The Senior Vice President, Nuclear Power, and the NSRB shall be notified within 24 hours.

2.2.5 A Licensee Event Report shall be prepared pursuant to 10 CFR 50.73. The Licensee Event Report shall be submitted to the Commission within 30 days of the violation.

2.2.6 Restart of the unit shall not commence until authorized by the Commission.

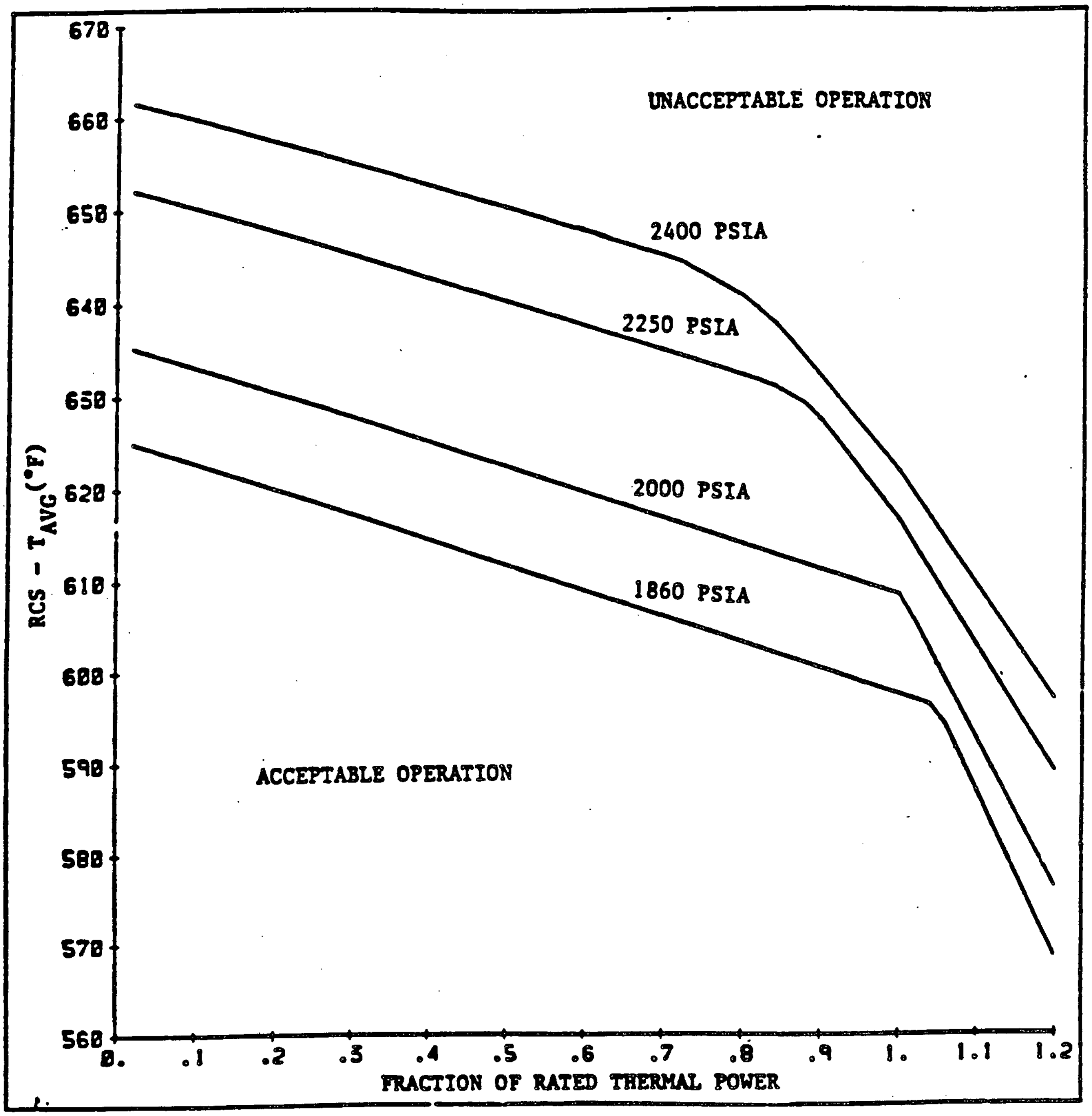


Figure 2.1.1-1 (Page 1 of 1)

Safety Limits

3.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION (LCO)

LCO 3.0.1 Limiting Conditions for Operation shall be met during the MODES or other specified conditions in the Applicability; except as provided in LCO 3.0.2.

LCO 3.0.2 Upon discovery of a failure to meet a Limiting Condition for Operation, the associated ACTIONS shall be met. If the Limiting Condition for Operation is restored prior to expiration of the specified Completion Time(s), completion of the Required Action is not required unless otherwise stated. Conditions in an LCO's ACTIONS may be concurrently applicable unless otherwise stated.

LCO 3.0.3 When an Limiting Condition for Operation is not met, and the associated ACTIONS are not met or an associated ACTION is not provided, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours,
b. MODE 4 within 13 hours, and
c. MODE 5 within 37 hours.

Exceptions to these requirements are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the Limiting Condition for Operation or ACTIONS, completion of the actions required by LCO 3.0.3 are not required.

LCO 3.0.3 is applicable in MODES 1, 2, 3 and 4.

(continued)

LIMITING CONDITIONS FOR OPERATION (LCO) (continued)

LCO 3.0.4	When a Limiting Condition for Operation is not met, entry into a MODE or other specified condition in the Applicability shall not be made unless 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 shall not prevent changes in MODES or other specified conditions in the Applicability which are required to comply with ACTIONS. Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow MODES or other specified conditions in the Applicability of the LCO to be entered while meeting the associated ACTIONS that do not permit continued operation in the MODE or other specified condition for an unlimited period of time.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS, may be returned to service under administrative control to perform testing required to demonstrate OPERABILITY, or the OPERABILITY of other equipment.

SURVEILLANCE REQUIREMENTS (SR)

SR 3.0.1	Each Surveillance Requirement shall be met during the MODES or other specified condition in the Applicability for its Limiting Conditions for Operation, unless otherwise stated in the individual Surveillance Requirement. Failure to meet a Surveillance Requirement shall be failure to meet the Limiting Condition for Operation. Surveillance Requirements Frequencies do not have to be met on inoperable equipment.
SR 3.0.2	The specified frequency for Surveillance Requirement 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 1.25 interval extension does not apply. If a Required Action requires performance of a Surveillance Requirement, or its Completion Time requires periodic performance of "once per...", the frequency extension above applies to the Completion Time interval. Exceptions to this requirement are stated in the individual Specifications.
SR 3.0.3	When a Limiting Condition for Operation is not met due to failure to perform a Surveillance within the specified frequency, the requirement to declare the equipment inoperable may be delayed for up to 24 hours from the time it is identified that the Surveillance has not been performed to permit the completion of the Surveillance. If the Surveillance is not performed within the 24 hour allowance, the Completion Time of the Required Actions begins immediately upon expiration of the 24 hour allowance. When the Surveillance is performed within the 24 hour allowance and the Surveillance Requirements are not met, the Completion Time of the Required Actions begins immediately upon the failure of the Surveillance.

(continued)

SURVEILLANCE REQUIREMENTS (SR) (continued)

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability shall not be made unless the Surveillance Requirements for the applicable LCO have been met. This provision shall not prevent passage through or to MODES or other specified conditions in compliance with Required Actions.

Exceptions to these requirements are stated in the individual Specifications, and allow 24 hours to complete the Surveillance beginning when the prerequisite conditions necessary to perform the Surveillance have been attained, unless otherwise stated.

3.1 REACTIVITY CONTROL

3.1.1 Shutdown Margin (T_{avg} > 200°F)

LCO 3.1.1 The SHUTDOWN MARGIN shall be $\geq$ to [1.6]% $\Delta k/k$.

APPLICABILITY: MODES 2 with k_{eff} < 1.0,
MODES 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SHUTDOWN MARGIN not within limit.	A.1 Initiate boration to restore SHUTDOWN MARGIN within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SHUTDOWN MARGIN within limit.	24 hours
SR 3.1.1.2	-------------------NOTE------------------- The SHUTDOWN MARGIN shall include an allowance for the withdrawn worth of rods that are immovable, as a result of mechanical interference or excessive friction, or rods that are untrippable. -- (continued)	-----NOTE---- Only required if one or more rods are inoperable, due to being immovable as a result of excessive friction or mechanical interference, or known to be untrippable ------------

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.1.2 (continued)	Demonstrate SHUTDOWN MARGIN is $\geq$ [1.6]% $\Delta k/k$	Within 1 hour AND 12 hours

CROSS-REFERENCES - None.

3.1 REACTIVITY CONTROL

3.1.2 <u>Shutdown Margin</u> ($T_{avg} \leq 200°F$)

LCO 3.1.2 The SHUTDOWN MARGIN shall be $\geq$ to [1.0]% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SHUTDOWN MARGIN not within limit.	A.1 Initiate boration to restore SHUTDOWN MARGIN within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.2.1	Demonstrate SHUTDOWN MARGIN is $\geq$ [1.0]% $\Delta k/k$	24 hours
SR 3.1.2.2	------------------NOTE------------------ The SHUTDOWN MARGIN shall include an allowance for the withdrawn worth of rods that are immovable, as a result of mechanical interference or excessive friction, or rods that are untrippable. -- (continued)	-----NOTE---- Only required if one or more rods are inoperable, due to being immovable as a result of excessive friction or mechanical interference, or known to be untrippable -------------

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.2.2 (continued)	Demonstrate SHUTDOWN MARGIN is ≥ [1.0]% Δk/k	Within 1 hour AND 12 hours

CROSS-REFERENCES - None.

3.1 REACTIVITY CONTROL

3.1.3 Core Reactivity

LCO 3.1.3 The Overall Core Reactivity Balance shall be within ± 1.0% $\Delta k/k$ of the predicted value.

APPLICABILITY: MODES 1 and 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limits.	A.1 Reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and surveillance requirements.	72 hours 72 hours
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Demonstrate core reactivity is within $\pm$ 1% $\Delta k/k$ of predicted values.	Prior to entering MODE 1 AND ----NOTE----- Only required after 60 EFPD ------------- 31 EFPD

CROSS-REFERENCES - None.

3.1 REACTIVITY CONTROL

3.1.4 Moderator Temperature Coefficient (MTC)

LCO 3.1.4 The MTC shall be within the limits specified in the COLR.

-------------------------NOTE-------------------------
The maximum positive limit shall be ≤ [0] Δk/k/°F at Hot Zero Power.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit,
MODES 1, 2, and 3 for the lower MTC limit.

-------------------------NOTE-------------------------
LCO 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	----------NOTE---------- Subsequent operation is permitted. The requirements of LCO 3.1.5, Control Bank Insertion Limits, remain applicable. ------------------------	
	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Rods not within withdrawal limits.	B.1 Restore rods to within withdrawal limits.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met.	C.1 Be in MODE 2 with $K_{eff} < 1.0$.	6 hours
D. MTC not within lower limit.	D.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
	------------------NOTE------------------- SR 3.0.4 is not applicable. --	
SR 3.1.4.1	Demonstrate MTC within upper limit.	After each refueling prior to entering MODE 1
SR 3.1.4.2	Demonstrate MTC within 300 ppm surveillance limit specified in the COLR.	-----NOTE---- Required once within 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm ------------- Each cycle

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	------------------NOTES------------------ 1. If the MTC is more negative than the 300 ppm surveillance limit provided in the COLR, SR 3.1.4.3 shall be repeated once per 14 Effective Full Power Days (EFPD) during the remainder of the fuel cycle. 2. SR 3.1.4.3 need not be repeated if the MTC measured at the equivalent of equilibrium RATED THERMAL POWER - All Rods Out (RTP-ARO) boron concentration of $\geq$ 60 ppm is less negative than the 60 ppm surveillance limit provided in the COLR. -- Demonstrate MTC within lower limit.	-----NOTE---- Required once within 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm ------------- Each cycle

CROSS-REFERENCES

TITLE	NUMBER
MODE 2 Physics Tests Exceptions	3.1.10
Startup Reports	5.10.2

3.1 REACTIVITY CONTROL

3.1.5 Rod Group Alignment Limits

LCO 3.1.5 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: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable.	A.1.1 Demonstrate SHUTDOWN MARGIN $\geq$[1.6]% $\Delta k/k$	1 hour
	OR	
	A.1.2 Initiate boration to restore SHUTDOWN MARGIN within limits.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1 Restore rod within alignment limits.	1 hour
	OR	
	B.2 ---------NOTE---------- Maintain bank sequence and insertion limits of LCO 3.1.4 and LCO 3.1.5, with changes to rod position or THERMAL POWER level, during subsequent operation. -----------------------	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	Realign remainder of rods in the group with the misaligned rod to within alignment limit.	1 hour
	<u>OR</u>	
	B.3.1.1 Verify SHUTDOWN MARGIN ≥ [1.6]% $\Delta k/k$.	1 hour
	<u>OR</u>	
	B.3.1.2 Initiate boration to restore SHUTDOWN MARGIN within limit.	1 hour
	<u>AND</u>	
	B.3.2 Reduce THERMAL POWER to ≤ 75% of RATED THERMAL POWER (RTP).	2 hours
	<u>AND</u>	
	B.3.3 Reduce Power Range Neutron Flux--High trip setpoints to ≤ 85% of RTP.	8 hours
	<u>AND</u>	
	B.3.4 Verify SHUTDOWN MARGIN ≥ [1.6]% $\Delta k/k$.	Once per 12 hours
	<u>AND</u>	
	B.3.5 Perform SR 3.2.1.1. ($F_Q(z)$ Surveillance).	72 hours
	<u>AND</u>	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.6 Perform SR 3.2.2.1. ($F_{\Delta H}^{N}$ Surveillance) **AND** B.3.7 Reevaluate safety analyses and confirm analyses results remain valid for duration of operation under these conditions.	72 hours 5 days
C. More than one rod not within alignment limit.	C.1 Be in MODE 3.	6 hours
D. One or more rod(s) aligned and trippable but immovable due to electrical problems in the Rod Control System.	D.1 ----------NOTE---------- Maintain bank sequence and insertions limits of LCO 3.1.4 and LCO 3.1.5, with changes to rod position or THERMAL POWER, during subsequent operation. ------------------------ Restore rods to OPERABLE status.	 72 hours
E. Required Actions and associated Completion Times for Condition B or D not met.	E.1 Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify individual rod positions within alignment limit as follows: a. With the rod position deviation monitor inoperable. <u>OR</u> b. With the rod position deviation monitor OPERABLE.	 4 hours 12 hours
SR 3.1.5.2	Move each rod not fully inserted in the core at least 10 steps in either direction.	92 days
SR 3.1.5.3	Demonstrate rod drop time of each rod, from the fully withdrawn position, is $\leq$ [2.4] seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry with: a. $T_{avg} \geq 500°F$, and b. All reactor coolant pumps operating.	Prior to reactor criticality after removal of the reactor head. <u>AND</u> 18 months

CROSS-REFERENCES

TITLE	NUMBER
Shutdown Bank Insertion Limit	3.1.6
Control Bank Insertion Limits	3.1.7
MODE 1 Physics Tests Exceptions	3.1.9
MODE 2 Physics Tests Exceptions	3.1.10
Axial Flux Difference	3.2.3
Quadrant Power Tilt Ratio	3.2.4

3.1 REACTIVITY CONTROL

3.1.6 Shutdown Bank Insertion Limits

LCO 3.1.6 Each shutdown bank shall be within its physical insertion limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$,
Within 15 minutes prior to initial control bank withdrawal during an approach to criticality.

---------------------------NOTE---------------------------
This LCO is not applicable while performing SR 3.1.3.2 (Rod Movement Test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limit(s).	A.1 Restore shutdown banks to within limit(s).	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	-------------------NOTE------------------ SR 3.0.4 is not applicable. --- Verify each shutdown bank within limit.	Within 15 minutes prior to initial control bank withdrawal during an approach to criticality <u>AND</u> 12 hours

CROSS-REFERENCES

TITLE	NUMBER
Rod Group Alignment Limits	3.1.5
Mode 1 Physics Tests Exceptions	3.1.9
Mode 2 Physics Tests Exceptions	3.1.10

3.1 REACTOR CONTROL

3.1.7 Control Bank Insertion Limits

LCO 3.1.7 Control banks shall be within the physical insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

---------------------------NOTE---------------------------
This LCO is not applicable while performing SR 3.1.5.2 (Rod Movement Test).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limit(s) not met.	A.1 Restore control bank(s) to within limit(s).	2 hours
B. Control bank(s) sequence or overlap limit(s) not met.	B.1 Restore control bank(s) sequence or overlap to meet limit(s)	2 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify predicted critical control bank position within limits.	Within 4 hours prior to achieving criticality
SR 3.1.7.2	-----------------NOTE------------------ SR 3.0.4 is not applicable. --------------------------------------- Verify each control bank insertion within limits as follows: a. With the rod insertion limit monitor inoperable. <u>OR</u> b. With the rod insertion limit monitor OPERABLE.	 4 hour 12 hours
SR 3.1.7.3	-----------------NOTE------------------ SR 3.0.4 is not applicable. --------------------------------------- Verify sequence and overlap limit(s) met for control banks not fully withdrawn from the core.	 12 hours

CROSS-REFERENCES

TITLE	NUMBER
Mode 1 Physics Tests Exceptions	3.1.9
Mode 2 Physics Tests Exceptions	3.1.10

3.1 REACTIVITY CONTROL

3.1.8 Rod Position Indication

LCO 3.1.8 The Analog Rod Position Indication System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One analog rod position indicator 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% of RATED THERMAL POWER (RTP).	8 hours
B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	8 hours
	<u>OR</u>		
	B.2	Reduce THERMAL POWER to $\leq$ 50% of RTP.	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One demand position indicator per bank inoperable for one or more banks.	C.1.1 Verify all analog rod position indicators for the affected banks are OPERABLE. AND C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are $\leq$ 12 steps apart. OR C.2 Reduce THERMAL POWER to $\leq$ 50% of RPT.	Once per 8 hours Once per 8 hours 8 hours
D. Required Actions and associated Completion Times not met	D.1 Be in MODE 3	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify each analog rod position indicator agrees within 12 steps of the group demand position for the full indicated range of rod travel.	12 hours

CROSS-REFERENCES - None.

3.1 REACTIVITY CONTROL

3.1.9 Mode 1 Physics Tests Exceptions

LCO 3.1.9 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.5 Rod Group Alignment Limits,
LCO 3.1.6 Shutdown Bank Insertion Limit,
LCO 3.1.7 Control Bank Insertion Limits,
LCO 3.2.3 Axial Flux Difference, and
LCO 3.2.4 Quadrant Power Tilt Ratio

may be suspended provided:

a. THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER (RTP), and

b. Power Range Neutron Flux--High trip setpoints are ≤ 10% of RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% of RTP.

APPLICABILITY: MODE 1 when performing PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Reduce THERMAL POWER to within limit. OR A.2 Suspend PHYSICS TESTS exceptions.	1 hour 1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Power Range Neutron Flux--High trip setpoints > 10% of RTP above the PHYSICS TESTS power level. OR Power Range Neutron Flux--High trip setpoints > 90% of RTP.	B.1 Restore Power Range Neutron Flux--High trip setpoints to ≤ 10% above the PHYSICS TESTS power level or to ≤ 90% of RTP, whichever is lower.	1 hour
	OR	
	B.2 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER ≤ 85% RTP.	1 hour
SR 3.1.9.2	Verify Power Range Neutron Flux--High trip setpoints within limit.	Within 8 hours prior to initiation of PHYSICS TESTS
SR 3.1.9.3	Perform SR 3.2.1.1 ($F_Q(z)$ Surveillance) and SR 3.2.2.1 ($F_{\Delta H}^N$ Surveillance).	12 hours

CROSS-REFERENCES

TITLE	NUMBER
Rod Group Alignment Limits	3.1.5
Shutdown Bank Insertion Limit	3.1.6
Control Bank Insertion Limits	3.1.7
Axial Flux Difference	3.2.3
Quadrant Power Tilt Ratio	3.2.4

3.1 REACTIVITY CONTROL

3.1.10 Mode 2 Physics Tests Exceptions

LCO 3.1.10 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4	Moderator Temperature Coefficient,
LCO 3.1.5	Rod Group Alignment Limits,
LCO 3.1.6	Shutdown Bank Insertion Limit,
LCO 3.1.7	Control Bank Insertion Limits, and
LCO 3.4.2	RCS Minimum Temperature for Criticality

may be suspended provided:

a. THERMAL POWER is maintained $\leq$ 5% of RATED THERMAL POWER, and

b. Reactor Coolant System (RCS) lowest loop average temperature is $\geq$ [541]°F.

APPLICABILITY: MODE 2 when performing PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open reactor trip breakers.	Immediately
B. RCS lowest loop average temperature not within limit.	B.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
	OR	
	B.2 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1	Perform an ANALOG CHANNEL OPERATIONAL TEST on each power range and intermediate range channel.	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.10.2	Verify the RCS lowest loop average temperature within limit.	30 minutes
SR 3.1.10.3	Verify THERMAL POWER within limit.	1 hour

CROSS-REFERENCES

TITLE	NUMBER
Moderator Temperature Coefficient	3.1.2
Rod Group Alignment Limits	3.1.3
Shutdown Bank Insertion Limit	3.1.4
Control Bank Insertion Limits	3.1.5
Reactor Trip System Instrumentation	3.3.1
RCS Minimum Temperature for Criticality	3.4.2

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$ (F_{XY} Methodology)

LCO 3.2.1 The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be maintained 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 at least 1% of RATED THERMAL POWER (RTP) for each 1% $F_Q(Z)$ exceeds limit.	15 minutes
	AND	
	A.2 Reduce AXIAL FLUX DIFFERENCE Acceptable Operation Limits as specified by the COLR.	4 hours
	AND	
	A.3 Reduce Power Range Neutron Flux--High trip setpoints at least 1% for each 1% $F_Q(Z)$ exceeds limit.	8 hours
	AND	
	A.4 Reduce Overpower ΔT trip setpoints at least 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours
	AND	
	(continued)	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of A.1
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
	-----NOTE----- SR. 3.0.4 is not applicable. --------------	
SR 3.2.1.1	Demonstrate corrected measured values $F_Q(Z)$ within limits.	31 Effective Full Power Days (EFPD) AND Prior to exeeding 75% of RTP after each refueling

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.1.2	------------NOTES (continued)------------ 1. If $F_{XY}^C > F_{XY}^L$, evaluate the effect of F_{XY} on the predicted $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit as specified by the COLR. 2. If $F_{XY}^{RTP} < F_{XY}^C \leq F_{XY}^L$, SR 3.2.1.2 shall be repeated within 24 hours after exceeding, by $\geq$ 20% of RTP, the THERMAL POWER at which F_{XY}^C was last determined. ---	
	Demonstrate $F_{XY}^C \leq F_{XY}^L$.	31 EFPD <u>AND</u> Prior to exceeding 75% of RTP after each refueling

CROSS-REFERENCES - None.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor-$F_{\Delta H}^{N}$

LCO 3.2.2 The Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{N}$, shall be maintained as specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------NOTE--------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ----------------------		
A. $F_{\Delta H}^{N}$ not within limit.	A.1.1 Restore $F_{\Delta H}^{N}$ to within limit. OR	4 hours
	A.1.2.1 Reduce THERMAL POWER to < 50% of RATED THERMAL POWER (RTP). AND (continued)	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1.2.2 Reduce Power Range Neutron Flux--High trip setpoints to ≤ 55% of RTP.	8 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.2.1.	24 hours
	<u>AND</u>	
	A.3 ---------NOTE---------- THERMAL POWER does not have to be reduced to comply with this Required Action. ----------------------	
	Perform SR 3.2.2.1.	Prior to exceeding 50% of RTP <u>AND</u> Prior to exceeding 75% of RTP <u>AND</u> 24 hours after reaching ≥ 95% of RTP
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
	-----------------NOTE------------------ SR 3.0.4 is not applicable. ---------------------------------------	
SR 3.2.2.1	Demonstrate $F_{\Delta H}^{N}$ within limits.	31 Effective Full Power Days AND Prior to exceeding 75% of RTP after each refueling

CROSS-REFERENCES - None.

3.2 POWER DISTRIBUTION LIMITS

3.2.3 Axial Flux Difference (CAOC Methodology)

LCO 3.2.3 The AXIAL FLUX DIFFERENCE (AFD):

a. Shall be maintained within the target band about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

------------------------NOTE--------------------------
The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
----

b. May deviate outside the target band with THERMAL POWER < 90% but ≥ 50% of RATED THERMAL POWER (RTP), provided AFD is within the Acceptable Operation Limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The Acceptable Operation Limits are specified in the COLR.

------------------------NOTE--------------------------
Penalty deviation time shall be accumulated on the basis of one minute penalty deviation for each minute of power operation with AFD outside the target band.
----

c. May deviate outside the target band with THERMAL POWER < 50% of RTP.

------------------------NOTE--------------------------
Penalty deviation time shall be accumulated on the basis of one-half minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% of RTP.

---------------------------NOTE--------------------------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time while performing surveillance testing of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within Acceptable Operation Limits.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. THERMAL POWER ≥ 90% of RTP. AND AFD not within the target band.	A.1	Restore AFD to within target band.	15 minutes
	OR		
	A.2	Reduce THERMAL POWER to < 90% of RTP.	15 minutes
----------NOTE--------- Required Actions B.1 and B.2 must be completed whenever Condition B is entered. ----------------------- B. Required Actions and associated Completion Times of Condition A not met.	B.1	Reduce THERMAL POWER to < 50% of RTP.	15 minutes
	AND		
	B.2	Reduce Power Range Neutron Flux--High trip setpoints to ≤ 55% of RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE--------- Required Actions C.1.1 and C.1.2, or C.2 must be completed whenever Condition C is entered. ----------------------		
THERMAL POWER < 90% ≥ 50% of RTP.	C.1.1 Reduce THERMAL POWER to < 50% of RTP.	30 minutes
AND	AND	
Cumulative penalty deviation time > 1 hour during previous 24 hours.	C.1.2 Reduce Power Range Neutron Flux--High trip setpoints to ≤ 55% of RTP.	8 hours
OR	OR	
AFD not within the target band and not within Acceptable Operation Limits.	C.2 Reduce THERMAL POWER to < 15% of RTP.	9 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.2.3.2	------------------NOTE------------------ Assume logged values of AFD exist during the preceding time interval. -- Verify AFD within limits and log AFD for each OPERABLE excore channel: a. With THERMAL POWER $\geq$ 90% of RTP. OR b. With THERMAL POWER > 15% < 90% of RTP.	-----NOTE---- Only required if AFD monitor alarm inoperable ------------- 15 minutes 1 hour
SR 3.2.3.3	------------------NOTE------------------ SR 3.0.4 is not applicable. -- Update target flux difference of each OPERABLE excore channel by: a. Determining the target flux difference in accordance with SR 3.2.3.4, or b. Using linear interpolation between the most recently measured value, and either the predicted value for the end of cycle or 0% AFD.	31 Effective Full Power Days (EFPD)
SR 3.2.3.4	------------------NOTE------------------ SR 3.0.4 is not applicable. -- Determine, by measurement, the target flux difference of each OPERABLE excore channel.	92 EFPD

CROSS-REFERENCES

TITLE	NUMBER
MODE 1 Physics Tests Exceptions	3.1.9

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Quadrant Power Tilt Ratio

LCO 3.2.4 The QUADRANT POWER TILT RATIO (QPTR) shall be $\leq$ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER (RTP).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER at least 3% from RTP for each 1% of QPTR > 1.00.	2 hours AND Evaluate QPTR and perform A.1 as necessary once per 12 hours thereafter
	AND	
	A.2 Reduce Power Range Neutron Flux--High trip setpoints at least 3% for each 1% of QPTR < 1.00.	8 hours AND Once per 12 hours thereafter
	AND	
	A.3 Perform SR 3.2.1.1 ($F_Q(Z)$ Surveillance) and SR 3.2.2.1 ($F_{\Delta H}^N$ Surveillance).	24 hours AND Once per 7 days thereafter
	AND	
	(continued)	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4.1 Reevaluate each analyses and confirm analyses results remain valid for duration of operation of operation under this condition at RTP. AND A.4.2 ----------NOTE--------- Only perform Required Action A.4.2 after A.4.1 is completed. ----------------------- Calibrate excore detectors to show zero QPTR.	Prior to increasing THERMAL POWER to RTP Prior to increasing THERMAL POWER to RTP
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER $\leq$ 50%.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify QPTR within limit by calculating QPTR as follows:	
	a. With QPTR alarm inoperable.	12 hours
	OR	
	b. With QPTR alarm OPERABLE.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	Verify QPTR within limit with the movable incore detectors by: a. Using two sets of 4 thimble locations with quarter-core symmetry, or b. Taking a power distribution flux map.	-----NOTE---- Only required if one power range channel is inoperable with THERMAL POWER < 75% of RTP ------------- 12 hours

CROSS-REFERENCES

TITLE	NUMBER
MODE 1 Physics Tests Exceptions	3.1.9

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System Instrumentation

LCO 3.3.1 The Reactor Trip System (RTS) instrumentation channels, trains, and interlocks, as shown in Table 3.3.1-1, shall be OPERABLE.

APPLICABILITY: According to Applicable Modes in Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels for one or more functions listed in Table 3.3.1-1 inoperable.	A.1 Enter the Condition(s) referenced in Table 3.3.1-1 for each inoperable function.	Immediately

(continued)

Table 3.3.1-1 (Page 1 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
1. Manual Reactor Trip	NA	NA	1/train, 2 trains	1,2 3[(a)],4[(a)],5[(a)]	B C	SR 3.3.1.15
2. Power Range, Neutron Flux						
A. High Setpoint	≤ 109% of RATED THERMAL POWER (RTP)	≤ 111.4% of RTP	4	1,2	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.18
B. Low Setpoint	≤ 25% of RTP	≤ 27.4% of RTP	4	1[(b)],2	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16 SR 3.3.1.18
C. f(ΔI)	Refer to Note 1 (Page 3.3-11)	Refer to Note 2 (Page 3.3-12)	4	1,2	E	SR 3.3.1.3 SR 3.3.1.6

(continued)

(a) With reactor trip breakers closed and Rod Control System capable of rod withdrawal.
(b) Below P-10 (Power Range Neutron Flux) setpoint.

Table 3.3.1-1 (Page 2 of 13)
Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
3. Power Range, Neutron Flux Rate						
A. High Positive Rate	≤ 5% of RTP with time constant ≥ 2 sec	≤ 6.3% of RTP with time constant ≥ 2 sec	4	1,2	D	SR 3.3.1.7 SR 3.3.1.12
B. High Negative Rate	≤ 5% of RTP with time constant ≥ 2 sec	≤ 6.3% of RTP with time constant ≥ 2 sec	4	1,2	D	SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.18
4. Intermediate Range Neutron Flux	≤ 25% of RTP	≤ 31.2% of RTP	2	2(c) 1(b),2(d)	F GH	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16

(continued)

(b) Below P-10 (Power Range Neutron Flux) setpoint.
(c) Below P-6 (Intermediate Range Neutron Flux) setpoint.
(d) Above P-6 (Intermediate Range Neutron Flux) setpoint.

Table 3.3.1-1 (Page 3 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
5. Source Range Neutron Flux	≤ 1E5 cps	≤ 1.4E5 cps				SR 3.3.1.1 SR 3.3.1.8
A. Startup			2	2(c)	I,J	SR 3.3.1.9
B. Shutdown			2	3(a),4(a),5(a)	K,J	SR 3.3.1.12
C. Shutdown			1	3(e),4(e),5(e)	L	SR 3.3.1.16
6. Overtemperature ΔT	Refer to Note 1 (Page 3.3-11)	Refer to Note 2 (Page 3.3-12)	4	1,2	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
7. Overpower ΔT	Refer to Note 3 (Page 3.3-13)	Refer to Note 4 (Page 3.3-14)	4	1,2	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18

(continued)

(a) With reactor trip breakers closed and Rod Control System capable of rod withdrawal.
(c) Below P-6 (Intermediate Range Neutron Flux) setpoint.
(e) With reactor trip breakers open.

Table 3.3.1-1 (Page 4 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
8. Pressurizer Pressure--Low	≥ 1970 psig	≥ 1952.4 psig	4	1(f)	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
9. Pressurizer Pressure--High	≤ 2385 psig	≤ 2402.6 psig	4	1,2	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
10. Pressurizer Water Level--High	≤ 92%	≤ 94.2%	3	1(f)	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11

(continued)

(f) Above P-7 (Low Power Reactor Trips Block) setpoint.

Table 3.3.1-1 (Page 5 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
11. Reactor Coolant Flow--Low						
A. Single Loop	≥ 90%	≥ 88.8%	3/loop	1(g)	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
B. Two Loops	≥ 90%	≥ 88.8%	3/loop	1(h)	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
12. Undervoltage-Reactor Coolant Pumps	≥ [4830] volts	≥ [4734] volts	[1]/bus	1(f)	E	SR 3.3.1.10(j) SR 3.3.1.11 SR 3.3.1.18
13. Underfrequency-Reactor Coolant Pumps	≥ [57] Hz	≥ [56.9] Hz	[1]/bus	1(f)	E	SR 3.3.1.10(j) SR 3.3.1.11 SR 3.3.1.18

(continued)

(f) Above P-7 (Low Power Reactor Trips Block) setpoint.
(g) Above P-8 (Power Range Neutron Flux) setpoint.
(h) Above P-7 (Low Power Reactor Trips Block) setpoint and below P-8 (Power Range Neutron Flux) setpoint.
(j) Final actuation not required during TRIP ACTUATION DEVICE OPERATIONAL TEST.

Table 3.3.1-1 (Page 6 of 13)

Reactor Trip System Instrumentation

	FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
14.	Steam Generator Water Level--Low-Low	See Note 5 (Page 3.3-14)	See Note 6 (Page 3.3-14)	3/steam generator	1,2	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.18
15.	Steam Generator Water Level--Low	See Note 5 (Page 3.3-14)	See Note 6 (Page 3.3-14)	1/steam generator	1,2	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11
	Coincident With						
	Steam/Feedwater Flow Mismatch	$\leq$ 38% of full steam flow at RTP	$\leq$ 41.9% of full steam flow at RTP	2/steam generator	1,2	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11

Table 3.3.1-1 (Page 7 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
16. Turbine Trip						
A. Low Fluid Oil Pressure	≥ [45] psig	≥ [43] psig	3	$1^{(i)}$	E	SR 3.3.1.11 SR 3.3.1.17
B. Turbine Stop Valve Closure	≥ [1%] open	≥ [1%] open	4	$1^{(i)}$	T	SR 3.3.1.11 SR 3.3.1.17
17. Safety Injection Input from Engineered Safety Features	NA	NA	2 trains	1,2	N	SR 3.3.1.15
18. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	≥ 1E-10 amps	≥ 6E-11 amps	2	$2^{(c)}$	P	SR 3.3.1.12 SR 3.3.1.13

(continued)

(c) Below P-6 (Intermediate Range Neutron Flux) setpoint.
(i) Above P-9 (Power Range Neutron Flux) setpoint.

Table 3.3.1-1 (Page 8 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
18. Reactor Trip System Interlocks (continued)						
B. Low Power Reactor Trips Block, P-7	NA	NA	1	1	P	SR 3.3.1.12 SR 3.3.1.14
P-10 Input	≤ 10% of RTP	≤ 12.4% of RTP	4	1	Q	SR 3.3.1.12 SR 3.3.1.14
<u>OR</u>						
P-13 Input	≤ 10% Turbine Power	≤ 12.4% Turbine Power	2	1	P	SR 3.3.1.11 SR 3.3.1.14
C. Power Range Neutron Flux, P-8	≤ 48% of RTP	≤ 50.4% of RTP	3	1	P	SR 3.3.1.12 SR 3.3.1.14
D. Power Range Neutron Flux, P-9	≤ 50% of RTP	≤ 52.4% of RTP	3	1	P	SR 3.3.1.12 SR 3.3.1.14
E. Power Range Neutron Flux, P-10	≥ 10% of RTP	≥ 7.6% of RTP	4	1(j) 1(b),2	P Q	SR 3.3.1.12 SR 3.3.1.14
F. Turbine Impulse Pressure, P-13	≤ 10% Turbine Power	≤ 12.4% Turbine Power	2	1	P	SR 3.3.1.11 SR 3.3.1.14

(b) Below P-10 (Power Range Neutron Flux) setpoint.

(j) Above P-10 (Power Range Neutron Flux) setpoint.

(continued)

Table 3.3.1-1 (Page 9 of 13)

Reactor Trip System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
19. Reactor Trip Breakers	NA	NA	2 trains	1,2............ 3(a),4(a),5(a)	R C	SR 3.3.1.4
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	NA	NA	1/Reactor Trip Breaker	1,2............ 3(a),4(a),5(a)	S C	SR 3.3.1.4
21. Automatic Trip Logic	NA	NA	2 trains	1,2............ 3(a),4(a),5(a)	R C	SR 3.3.1.5

(a) With reactor trip breakers closed and Rod Control System capable of rod withdrawal.

Table 3.3.1-1 (Page 10 of 13)
Reactor Trip System Instrumentation
Table Notations

Note 1: Overtemperature ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT
$\frac{(1+\tau_1 S)}{(1+\tau_2 S)}$	=	Lead-lag compensator on measured ΔT
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 seconds, τ_2 = 3 seconds
$\left(\frac{1}{1+\tau_3 S}\right)$	=	Lag compensator on measured ΔT
τ_3	=	Time constant utilized in lag compensator for ΔT, τ_3 = 0 seconds
ΔT_0	=	Indicated ΔT at RTP
K_1	=	1.0952
K_2	=	0.01327
$\frac{(1+\tau_4 S)}{(1+\tau_5 S)}$	=	The function generated by lead-lag compensator for T_{avg} dynamic compensation
τ_4, τ_5	=	Time constants utilized in lead-lag compensator for T_{avg}, τ_4 = 33 seconds, τ_5 = 4 seconds
T	=	Measured average temperature, °F
$\left(\frac{1}{1+\tau_6 S}\right)$	=	Lag compensator on measured T_{avg}
τ_6	=	Time constant utilized in measured T_{avg} lag compensator, τ_6 = 0 seconds

(continued)

Table 3.3.1-1 (Page 11 of 13)
Reactor Trip System Instrumentation
Table Notations

<u>Note 1: Overtemperature T (continued)</u>

$T' \leq$ 588.2 °F (NOMINAL T_{avg} at RATED THERMAL POWER)

K_3 = 0.0006476

P = Measured pressurizer pressure, psig

P′ = 2235 psig (Nominal Reactor Coolant System operating pressure)

S = Laplace transform operator, sec^{-1}, and

f_1 (ΔI) is a function of the indicated difference between the upper and lower detectors of the power range neutron detectors, with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -32% and +10%, f_1 (ΔI)= 0, where q_t and q_b are percent RTP in the upper and lower halves of the core respectively, and $q_t + q_b$ is the total THERMAL POWER in percent of RTP,

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -32%, the ΔT Trip Setpoint shall be automatically reduced by 1.34% of its value at RTP, and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 1.22% of its value at RTP.

<u>Note 2</u>: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0% of ΔT span.

Reactor Trip System Instrumentation
Table Notations

<u>Note 3: Overpower ΔT</u>

$$\Delta T \; \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{ K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \frac{1}{1 + \tau_6 S} - T'' \right] - f_2(\Delta I) \right\}$$

Where:

ΔT	=	Measured ΔT
$\frac{(1 + \tau_1 S)}{(1 + \tau_2 S)}$	=	Lead-lag compensator on measured T
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 12$ seconds, $\tau_2 \leq 3$ seconds
$\left(\frac{1}{1 + \tau_3 S}\right)$	=	Lag compensator on measured ΔT
τ_3	=	Time constant utilized in lag compensator for ΔT, $\tau_3 = 2$ seconds
ΔT_0	=	Indicated ΔT at RTP
K_4	$\leq$	1.09
K_5	$\geq$	0.02 / °F for increasing average temperature and 0.0 for decreasing average temperature
$\left(\frac{\tau_7 S}{1 + \tau_7 S}\right)$	=	The function generated by rate-lag compensator for T_{avg} dynamic compensation
τ_7	=	Time constant utilized in rate-lag compensator for T_{avg}, $\tau_7 = 5$ seconds
$\left(\frac{1}{1 + \tau_6}\right)$	=	Lag compensator on measured T_{avg}
τ_6	=	Time constant utilized in measured T_{avg} lag compensator, $\tau_6 = 0$ seconds

(continued)

Table 3.3.1-1 (Page 13 of 13)
Reactor Trip System Instrumentation
Table Notations

<u>Note 3: Overpower T (continued)</u>

K_6 $\geq$ 0.00128 / °F for T > T'' and K_6 = 0 for T $\leq$ T''

T = Measured average temperature, °F

T'' = Nominal T_{avg} at RTP (calibration temperature for ΔT instrumentation, $\leq$ 588.2 °F)

S = Laplace transform operator, sec^{-1}, and

$f_2(\Delta I)$ = 0 for all ΔI

<u>Note 4</u>: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than [0.9]% of ΔT span.

<u>Note 5:</u> A setpoint $\geq$ 17% between 0 and 30% load, and increasing linearly to $\geq$ 54.9% at 100% load.

<u>Note 6:</u> A setpoint $\geq$ 14.8% between 0 and 30% load, and increasing linearly to $\geq$ 52.7% at 100% load.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Open reactor trip breakers.	55 hours
C. One channel inoperable.	C.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2 Open reactor trip breakers.	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	----------NOTE--------- The channel may be bypassed for up to 4 hours for surveillance testing of other channels and resetting Trip Setpoint. -----------------------	
	D.1.1 Reduce THERMAL POWER to $\leq$ 75 % of RTP.	4 hours
	AND	
	D.1.2 Place channel in tripped condition.	6 hours
	AND	
	D.1.3 Reduce Power Range, Neutron Flux--High Setpoint trip to $\leq$ 85% of RTP.	8 hours
	OR	
	D.2.1 Place channel in tripped condition.	6 hours
	AND	
	D.2.2 Perform SR 3.2.4.2, Calculation of Quadrant Power Tilt Ratio.	Once per 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	----------NOTE---------- The channel may be bypassed for up to 4 hours for surveillance testing of other channels. ------------------------	
	E.1 Place channel in tripped condition.	6 hours
F. THERMAL POWER below P-6, one or two channels inoperable.	F.1 Restore channels to OPERABLE status.	Prior to increasing THERMAL POWER above P-6
G. THERMAL POWER above P-6, but below 10% of RTP, one channel inoperable.	G.1 Reduce THERMAL POWER to < the P-6 setpoint.	2 hours
	<u>OR</u>	
	G.2 Increase THERMAL POWER to ≥ 10% of RTP (P-10 setpoint).	2 hours
H. THERMAL POWER above P-6, but below 10% of RTP, two channels inoperable.	H.1 Suspend operations involving positive reactivity additions.	15 minutes
	<u>AND</u>	
	H.2 Reduce THERMAL POWER to < the P-6 setpoint.	2 hours
I. One channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	15 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. Two channels inoperable.	J.1 Suspend operations involving positive reactivity additions.	15 minutes
	<u>AND</u>	
	J.2 Open reactor trip breakers.	15 minutes
K. One channel inoperable.	K.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	K.2.1 Open reactor trip breakers.	49 hours
	<u>AND</u>	
	K.2.2 Suspend operations involving positive reactivity additions.	49 hours
	<u>AND</u>	
	K.2.3 Perform Required Action A.2 of LCO 3.9.2, secure unborated water source isolation valves.	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. No channels OPERABLE.	L.1 Suspend operations involving positive reactivity additions.	15 minutes
	<u>AND</u>	
	L.2 Perform Required Action A.2 of LCO 3.9.2, secure unborated water source isolation valves.	1 hour
	<u>AND</u>	
	L.3 Perform SR 3.1.1.1, calculate Shutdown Margin.	Within 1 hour <u>AND</u> Once per 12 hours thereafter
M. One channel inoperable.	M.1 Place channel in tripped condition.	6 hours
N. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 2 hours for surveillance testing. ------------------------	
	N.1 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
P. One channel inoperable.	P.1 Verify interlock is in required state for existing plant conditions.	1 hour
Q. Two-out-of-four channels inoperable.	Q.1 Verify interlock is in required state for existing plant conditions.	Within 15 minutes after going below P-10 setpoint
R. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 2 hours for surveillance testing or maintenance on undervoltage or shunt trip mechanisms. ------------------------	
	R.1 Be in MODE 3.	6 hours
S. Reactor trip breaker undervoltage or shunt trip mechanism inoperable.	S.1 Declare reactor trip breaker inoperable.	48 hours
T. One channel inoperable.	T.1 Restore to OPERABLE status.	48 hours
	<u>OR</u>	
	T.2 Reduce THERMAL POWER < P-9.	54 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	--------------------NOTES------------------ 1. The CHANNEL CALIBRATION shall consist only of a comparison of Nuclear Instrumentation System (NIS) channel with results of the calorimetric. 2. Adjust NIS channel if absolute difference is > 2%. --- Perform CHANNEL CALIBRATION.	-----NOTE----- Only required if THERMAL POWER > 15% of RTP -------------- 24 hours
SR 3.3.1.3	--------------------NOTES------------------ 1. The CHANNEL CALIBRATION shall consist only of a comparison of NIS AXIAL FLUX DIFFERENCE with results of the the results of the incore system. 2. Recalibrate NIS channel if absolute difference is > 3%. --- Perform CHANNEL CALIBRATION.	-----NOTE----- Only required if THERMAL POWER > 15% of RTP -------------- 31 Effective Full Power Days (EFPD)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.4	--------------------NOTES------------------ 1. This test shall include independent verification of reactor trip breaker undervoltage and shunt trip mechanisms. 2. The shunt test must be performed on reactor trip bypass breaker prior to placing the bypass breaker in service. -- Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	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------------------ The CHANNEL CALIBRATION shall consist only of an incore-excore calibration. -- Perform CHANNEL CALIBRATION.	-----NOTE----- Only required if THERMAL POWER > 50% of RTP -------------- 92 EFPD
SR 3.3.1.7	Perform ANALOG CHANNEL OPERATIONAL TEST.	23 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	--------------------NOTE------------------- This test shall include verification that interlocks P-6 and P-10 are in their required state for existing plant conditions. --- Perform ANALOG CHANNEL OPERATIONAL TEST.	-----NOTE----- Only required in MODES 3, 4, or 5 when the Rod Control System is capable of rod withdrawal -------------- 92 days
SR 3.3.1.9	--------------------NOTE------------------- This test shall include verification that High Flux at Shutdown alarm setpoint $\leq$ one-half decade above background. --- Perform ANALOG CHANNEL OPERATIONAL TEST.	-----NOTE----- Only required in MODES 3, 4, or 5 -------------- 92 DAYS
SR 3.3.1.10	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.11	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	--------------------NOTE------------------- The neutron detectors may be excluded from the CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.13	Perform ANALOG CHANNEL OPERATIONAL TEST.	18 months
SR 3.3.1.14	-------------------NOTE------------------ This test shall consist of verification that the interlock is in required state with power level $\geq$ the interlock Trip Setpoint. --- Perform ANALOG CHANNEL OPERATIONAL TEST.	18 months
SR 3.3.1.15	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	18 months
SR 3.3.1.16	Perform ANALOG CHANNEL OPERATIONAL TEST.	-----NOTE----- Only required if not performed within previous 31 days -------------- Prior to reactor startup

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.17	--------------------NOTE------------------- Verification of Trip Setpoint is not required. --	-----NOTE----- Only required if not performed within previous 31 days -------------
	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	Prior to reactor startup
SR 3.3.1.18	Perform REACTOR TRIP SYSTEM RESPONSE TIME test.	18 months on a STAGGERED TEST BASIS

CROSS-REFERENCES

TITLE		NUMBER
Reactor Trip System Table 3.3.1-1	ESFAS Table 3.3.2-1	3.3.2
Function 6. Overtemperature ΔT	Function 1.D Pressurizer Pressure--Low Function 1.F High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident With Steam Line Pressure--Low Function 10.B interlock P-11 Pressurizer Pressure Function 10.C interlock P-12 T_{avg}--Low-Low	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Reactor Trip System Table 3.3.1-1	ESFAS Table 3.3.2-1	3.3.2
Function 7. Overpower ΔT	Function 1.F High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident With Steam Line Pressure--Low Function 10.C interlock P-12 Tavg--Low-Low	
Function 8. Pressurizer Pressure--Low	Function 1.D Pressurizer Pressure--Low Function 10.B interlock P-11 Pressurizer Pressure	
Function 9. Pressurizer Pressure--High	Function 1.D Pressurizer Pressure--Low Function 10.B interlock P-11 Pressurizer Pressure	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Reactor Trip System Table 3.3.1-1	ESFAS Table 3.3.2-1	3.3.2
Function 14. Steam Generator Water Level--Low-Low	Function 5.B Steam Generator Water Level--High-High Function 6.B Steam Generator Water Level--Low-Low Function 10.D interlock P-14 Steam Generator Water Level-High-High	
Function 15. Steam Generator Level-Low Coincident With Steam/Feedwater Flow Mismatch	Function 1.E Steam Line Pressure--Low Function 1.F High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low OR Coincident with Steam Line Pressure--low Function 5.B Steam Generator Water Level--High-High Function 6:B Steam Generator Water Level--Low-Low Function 10.D interlock P-14 Steam Generator Water Level--High-High	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Reactor Trip System Table 3.3.1-1	ESFAS Table 3.3.2-1	3.3.2
Function 19. Reactor Trip Breakers	Function 10.A interlock P-4 Reactor Trip	
Reactor Trip System Table 3.3.1-1	Accident Monitoring Instrumentation Table 3.3.3-1	3.3.3
Function 6. Overtemperature ΔT	Function 3. Reactor Coolant Outlet Temperature - Thot (Wide Range) Function 4. Reactor Coolant Inlet Temperature - Tcold (Wide Range)	
Function 7. Overpower ΔT	Function 3. Reactor Coolant Outlet Temperature - Thot (Wide Range) Function 4. Reactor Coolant Inlet Temperature - Tcold (Wide Range)	
Function 10. Pressurizer Water Level--High	Function 1. Pressurizer Water Level	
Function 14. Steam Generator Water Level--Low-Low	Function 14. Steam Generator Water Level (Narrow Range)	
Function 16. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	Function 14 Steam Generator Water Level (Narrow Range) Function 16 Steam Line Pressure	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
<u>Reactor Trip System Table 3.3.1-1</u>	<u>Remote Shutdown System Instrumentation Table 3.3.4-1</u>	3.3.4
Function 5. Source Range Neutron Flux	Function 4. Source Range Neutron Flux	
Function 6. Overtemperature ΔT	Function 3. Reactor Coolant Temperature - Hot Leg	
Function 7. Overpower ΔT	Function 3. Reactor Coolant Temperature - Hot Leg	
Function 10. Pressurizer Water Level--High	Function 1. Pressurizer Water Level	
Function 14. Steam Generator Water Level--Low-Low	Function 10. Steam Generator Water Level	
Function 14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	Function 9. Steam Generator Pressure Function 10. Steam Generator Water Level	
Function 19. Reactor Trip Breakers	Function 5. Reactor Trip Breaker Open/Closed Indication	

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Features Actuation System Instrumentation

LCO 3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels, trains, and interlocks, as shown in Table 3.3.2-1, shall be OPERABLE.

APPLICABILITY: According to Applicable Modes in Table 3.3.2-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels for one or more functions listed in Table 3.3.2-1 inoperable.	A.1 Enter the Condition(s) referenced in Table 3.3.2.1 for each inoperable function.	Immediately

(continued)

Table 3.3.2-1 (Page 1 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
1. SAFETY INJECTION						
A. Manual Initiation	N/A	N/A	1/train, 2 trains	1,2,3,4	B	SR 3.3.2.7
B. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
C. Containment Pressure--High	≤ 1.54 psig	≤ 1.8 psig	3	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
D. Pressurizer Pressure--Low	≥ 1870 psig	≥ 1852.4 psig	3	1,2,3[(a)]	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
E. Steam Line Differential Pressure	≤ 100 psig	≤ 124.0 psig	3/steam line	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

(continued)

(a) This function may be blocked in this MODE below the P-11 (Pressurizer Pressure) setpoint.

Table 3.3.2-1 (Page 2 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
1. <u>SAFETY INJECTION</u> (continued)						
F. High Steam Flow in Two Steam Lines	See Note 1 (Page 3.3-46)	See Note 2 (Page 3.3-46)	2/steam line	1,2,3[b]	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
Coincident With						
T_{avg}--Low-Low	≥ 550 °F	≥ 549.2 °F	1/loop	1,2,3[b]	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
<u>OR</u>						
Coincident With						
Steam Line Pressure--Low	≥ 675 psig[c]	≥ 646.4 psig[c]	1/steam line	1,2,3[b]	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

(continued)

(c) Time constants utilized in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

(b) This function may be blocked in this MODE below the P-12 (T_{avg}--Low-Low) setpoint.

Table 3.3.2-1 (Page 3 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
2. <u>CONTAINMENT SPRAY</u>						
A. Manual Initiation	N/A	N/A	1 pair/train, 2 trains	1,2,3,4	B	SR 3.3.2.7
B. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
C. Containment Pressure--High-High	≤ 2.81 psig	≤ 3.1 psig	4	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

(continued)

Table 3.3.2-1 (Page 4 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
3. <u>CONTAINMENT ISOLATION</u>						
A. Phase "A" Isolation						
1. Manual Initiation	N/A	N/A	1/train, 2 trains	1,2,3,4	B	SR 3.3.2.7
2. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
3. Safety Injection	Refer to Function 1 (SAFETY INJECTION) for all initiating functions and requirements.					
B. Phase "B" Isolation						
1. Manual Initiation	N/A	N/A	1 pair/train, 2 trains	1,2,3,4	B	SR 3.3.2.7
2. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8

(continued)

Table 3.3.2-1 (Page 5 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
3. <u>CONTAINMENT ISOLATION</u>						
B. Phase "B" Isolation (continued)						
3. Containment Pressure--High-High	≤ 2.81 psig	≤ 3.1 psig	4	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
C. Containment Vent Isolation						
1. Manual Initiation	N/A	N/A	1/train, 2 trains	1,2,3,4	E	SR 3.3.2.7
2. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	E	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
3. Safety Injection	Refer to Function 1 (SAFETY INJECTION) for all initiating functions and other requirements.				E	
4. Containment Radiation-High						
a. Gaseous Monitor	≤ [2] times background	N/A	1	1,2,3,4	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6

(continued)

Table 3.3.2-1 (Page 6 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
4. Containment Radiation-High (continued)						
b. Particulate Monitor	≤ [2] times background	N/A	1	1,2,3,4	F	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6
4. <u>STEAM LINE ISOLATION</u>						
A. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
B. Containment Pressure--High-High	≤ 2.81 psig	≤ 3.1 psig	4	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

(continued)

Table 3.3.2-1 (Page 7 of 14)

Engineered Safety Features Actuation System

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
4. <u>STEAM LINE ISOLATION</u> (continued)						
c. High Steam Flow in Two Steam Lines	See Note 1 (Page 3.3-46)	See Note 2 (Page 3.3-46)	2/steam line	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
Coincident With						
T_{avg}--Low-Low	≥ 550°F	≥ 549.2°F	1/loop	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
<u>OR</u>						
Coincident With						
Steam Line Pressure--Low	≥ 675 psig(b)	≥ 646.4 psig(b)	1/steam line	1,2,3	D	SR 3.2.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

(continued)

(b) Time constants utilized in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

Table 3.3.2-1 (Page 8 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
5. <u>TURBINE TRIP AND FEEDWATER ISOLATION</u>						
A. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
B. Steam Generator Water Level--High-High	≤ 82.4% narrow range	≤ 84.6% narrow range	3/steam generator	1,2,3	D	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.8
C. Safety Injection	Refer to Function 1 (SAFETY INJECTION) for all initiating functions and requirements.					

(continued)

Table 3.3.2-1 (Page 9 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
6. <u>AUXILIARY FEEDWATER</u>						
A. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8
B. Steam Generator Water Level--Low-Low	See Note 3 (Page 3.3-46)	See Note 4 (Page 3.3-46)	3/steam generator	1,2,3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
C. Safety Injection	Refer to Function 1 (SAFETY INJECTION) for all initiating functions and requirements.					
D. Loss of Offsite Power	[4830] v 0.0 volt input to inverse time relay with a 5 second time delay.	[4830]±96.6 v 0.0 volt input to inverse time relay with a 5 ± 1 second time delay.	2/bus	1,2,3	H	SR 3.3.2.5(d) SR 3.3.2.6 SR 3.3.2.8
E. Trip of All Main Feedwater Pumps	N/A	N/A	1/pump	1,2	H	SR 3.3.2.7 SR 3.3.2.8

(d) Testing shall consist of relay testing excluding final actuation of the pumps or valves.

(continued)

Table 3.3.2-1 (Page 10 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
6. AUXILIARY FEEDWATER (continued)						
F. Auxiliary Feedwater Pump Suction Pressure--Low						
1. Supply Valve for Motor-	≥ [2.15] psig	≥ [1.65] psig	3/pump	1,2,3	B	SR 3.3.2.4(e) SR 3.3.2.6 SR 3.3.2.8
2. Supply valve for Turbine	≥ [13.1] psig	≥ [12.1] psig	3/pump	1,2,3	B	SR 3.3.2.4(e) SR 3.3.2.6 SR 3.3.2.8
7. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP						
A. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8

(continued)

(e) Final actuation of the valves not required.

Table 3.3.2-1 (Page 11 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
7. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP (continued)						
B. RWST Level--Low-Low	≥[130] in. from tank base	≥[126] in. from tank base	4	1,2,3,4	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
Coincident With						
Containment Sump Level--High	≤ [30] in. above El. 703 feet	≤ [32.5] in. above El. 703 feet	4	1,2,3,4	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6
AND						
Coincident With						
Safety Injection	Refer to Function 1 (SAFETY INJECTION) for all initiating functions and requirements.					

(continued)

Table 3.3.2-1 (Page 12 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
8. CONTROL ROOM EMERGENCY VENTILATION						
A. Manual Initiation	N/A	N/A	1/train, 2 trains	All	I	SR 3.3.2.7
B. Automatic Actuation Logic and Actuation Relays	N/A	N/A	2 trains	1,2,3,4	I	SR 3.3.2.2 SR 3.3.2.3
C. Phase "A" Isolation	Refer to Function 3.A (Phase "A" Isolation) for all initiating functions and other requirements.				I	
D. Area Radiation	≤[400] cpm	N/A	2	ALL	J	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6

(continued)

Table 3.3.2-1 (Page 13 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
9. <u>LOSS OF POWER</u>						
A. Loss of Voltage						
1. Diesel Generator Start	[4830] v 0.0 volt input to inverse time relay with a 1.5 second time delay.	[4830]±96.6 v 0.0 volt input to inverse time relay with a 1.5 ± 1 second time delay.	2/bus	1,2,3,4	D	SR 3.3.2.5(f) SR 3.3.2.6
2. Load Shedding	[4830] v 0.0 volt input to inverse time relay with a 5 second time delay.	[4830]±96.6 v 0.0 volt input to inverse time relay with a 5 ± 1 second time delay.	2/bus	1,2,3,4	D	SR 3.3.2.5(f) SR 3.3.2.6
B. Degraded Voltage						
1. Diesel Generator Start and Load Shedding	[6560] v with a time delay [300] seconds	[6560]± 33 v with a time delay [300] ± 30 seconds	2/bus	1,2,3,4	D	SR 3.3.2.5(f) SR 3.3.2.6
2. Safety Injection Diesel Generator Start and Load Shedding	[6560] Volts with a time delay [10] seconds	[6560]± 33 v with a time delay [10] ± 1 second	2/bus	1,2,3,4	D	SR 3.3.2.5(f) SR 3.3.2.6

(f) Testing shall consist of voltage sensor relay testing excluding actuation of load shedding, diesel generator start, and time delay timers.

(continued)

Table 3.3.2-1 (Page 14 of 14)

Engineered Safety Features Actuation System Instrumentation

FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	CONDITION	SURVEILLANCE REQUIREMENTS
10. <u>ESFAS INTERLOCKS</u>						
A. Reactor Trip, P-4	N/A	N/A	2/train, 2 trains	1,2,3	H	SR 3.3.2.7
B. Pressurizer Pressure, P-11	≤ 1970 psig	≤ 1987.6 psig	3	1,2,3	K	SR 3.3.2.4 SR 3.3.2.6
C. T_{avg}--Low-Low, P-12	≥550°f	≤550.8°F ≥549.2°F	1/loop	1,2,3	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8
D. Steam Generator Water Level--High-High, P-14	≤82.4% narrow range	≤84.6% narrow range	3/steam generator	1,2,3	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.8

Notation for Table 3.3.2-1

Note 1: ≤ a function defined as follows: P corresponding to 40% of full steam flow at 0% load and then a P increasing linearly to a P corresponding to 110% of full steam flow at full load.

Note 2: ≤ a function defined as follows: P corresponding to 44.6% of full steam flow at 0% load and then a P increasing linearly to a P corresponding to 114.6% of full steam flow at full load.

Note 3: ≥ 17% between 0 and 30% load, increasing linearly to ≥54.9% at 100% nominal load.

Note 4: ≥ 14.8% between 0 and 30% load, increasing linearly to ≥52.7% at 100% nominal load.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One train or channel inoperable.	B.1 Restore train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Be in MODE 5.	84 hours
C. One train inoperable.	---------NOTE---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------	
	C.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	C.2 Be in MODE 5.	42 hours
D. One channel inoperable.	----------NOTE--------- The channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------	
	D.1 Place channel in tripped condition.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One train inoperable.	E.1 Place and maintain containment purge and exhaust valves in closed position.	4 hours
F. One or more required purge and exhaust radiation channel setpoints outside limits.		------NOTE----- Completion Time on a per channel basis ----------------
	F.1 Adjust setpoint to within limits.	4 hours
	<u>OR</u>	
	F.2 Place and maintain containment purge and exhaust valves in closed position.	4 hours
G. One train inoperable.	--------NOTE----------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -----------------------	
	G.1 Be in MODE 3	12 hours
	<u>AND</u>	
	G.2 Be in MODE 4	18 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. One train inoperable.	H.1 Restore train to OPERABLE status. <u>OR</u> H.2.1 Be in MODE 3. <u>AND</u> H.2.2 Be in MODE 4.	48 hours 54 hours 60 hours
I. One train inoperable.	I.1 Place Control Room Emergency Ventilation System in isolation mode of operation.	7 days
J. One channel inoperable.	J.1 Place Control Room Emergency Ventilation System in isolation mode of operation.	1 hour
K. One channel inoperable.	K.1 Verify interlock is in required state for existing plant condition.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.3	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.4	Perform ANALOG CHANNEL OPERATIONAL TEST.	92 days
SR 3.3.2.5	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	92 days
SR 3.3.2.6	-------------------NOTE------------------- This test shall include verification that the time constants are adjusted to the prescribed values. --------------------------------------- Perform CHANNEL CALIBRATION.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.7	Perform TRIP ACTUATING DEVICE OPERATIONAL TEST.	18 months
SR 3.3.2.8	Perform ENGINEERED SAFETY FEATURE RESPONSE TIME test.	18 months on a STAGGERED TEST BASIS

CROSS-REFERENCES

TITLE		NUMBER
<u>ESFAS</u> <u>Table 3.3.2-1</u>	<u>Reactor Trip System</u> <u>Table 3.3.1-1</u>	3.3.1
Function 1.D Pressurizer Pressure--Low	Function 6. Overtemperature ΔT Function 8. Pressurizer Pressure--Low Function 9. Pressurizer Pressure--High	
Function 1.E Steam Line Pressure--Low	Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 6. Overtemperature ΔT Function 7. Overpower ΔT Function 15. Level--Low Coincident With Steam/Feedwater Flow Mismatch	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 6. Overtemperature ΔT Function 7. Overpower ΔT Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
<u>ESFAS</u> <u>Table 3.3.2-1</u>	<u>Reactor Trip System</u> <u>Table 3.3.1-1</u>	3.3.1
Function 5.B Steam Generator Water Level--High-High	Function 14. Steam Generator Water Level--Low-Low Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
Function 6.B Steam Generator Water Level--Low-Low	Function 14. Steam Generator Water Level--Low-Low Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
Function 10.A interlock P-4 Reactor Trip	Function 19. Reactor Trip Breakers	
Function 10.B interlock P-11 Pressurizer Pressure--Low	Function 6. Overtemperature ΔT Function 8. Pressurizer Pressure--Low Function 9. Pressurizer Pressure--Low	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Function 10.C interlock P-12 Tavg--Low-Low	Function 6. Overtemperature ΔT Function 7. Overpower ΔT	
<u>ESFAS</u> <u>Table 3.3.2-1</u>	<u>Reactor Trip System</u> <u>Table 3.3.1-1</u>	3.3.1
Function 10.D interlock P-14 Steam Generator Water Level--High-High	Function 14. Steam Generator Water Level--Low-Low Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
<u>ESFAS</u> <u>Table 3.3.2-1</u>	<u>Accident Monitoring</u> <u>Table 3.3.3-1</u>	3.3.1
Function 1.C Containment Pressure--High	Function 11. Containment Pressure	
Function 1.E Steam Line Differential Pressure	Function 16. Steam Line Pressure	
Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 3. Reactor Coolant Outlet Temperature - T hot (Wide Range) Function 4. Reactor Coolant Inlet Temperature - T cold (Wide Range) Function 16. Steam Line Pressure	
Function 2.C Containment Pressure-- High-High	Function 11. Containment Pressure	
Function 3.B.3 Containment Pressure-- High-High	Function 11. Containment Pressure	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
ESFAS Table 3.3.2-1	Accident Monitoring Instrumentation Table 3.3.3-1	3.3.3
Function 4.B Containment Pressure--High-High	Function 11. Containment Pressure	
Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 3. Reactor Coolant Outlet Temperature - T hot (Wide Range) Function 4. Reactor Coolant Inlet Temperature - T cold Function 16. Steam Line Pressure	
Function 5.B Steam Generator Water Level--High-High	Function 14. Steam Generator Water Level (Narrow Range)	
Function 6.B Steam Generator Water Level--Low-Low	Function 14. Steam Generator Water Level (Narrow Range)	
Function 7.B Refueling Water Storage Tank (RWST) Level--Low-Low Coincident With Containment Sump Level--High AND Coincident With Safety Injection	Function 12. Containment Sump Water Level Function 13. Refueling Water Storage Tank Level	
Function 10.C interlock P-12 Tavg--Low-Low	Function 3. Reactor Coolent Outlet Temperature - T hot (Wide Range) Function 4. Reactor Coolant Inlet Temperature - T cold (Wide Range)	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
ESFAS Table 3.3.2-1	Accident Monitoring Instrumentation Table 3.3.3-1	3.3.3
Function 10.D interlock P-14 Steam Generator Water Level--High-High	Function 14. Steam Generator Water Level (Narrow Range)	
ESFAS Table 3.3.2-1	Remote Shutdown System Table 3.3.4-1	3.3.4
Function 1.C Containment Pressure--High	Function 7. Containment Pressure	
Function 1.E Steam Line Differential Pressure	Function 9. Steam Generator Pressure	
Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 3. Reactor Coolant Outlet Temperature - Hot Leg Function 9. Steam Generator Pressure	
Function 2.C Containment Pressure--High	Function 7. Containment Pressure	
Function 3.B.3 Containment Pressure--High-High	Function 7. Containment Pressure	
Function 4.B Containment Pressure--High-High	Function 7. Containment Pressure	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
ESFAS Table 3.3.2-1	Remote Shutdown System Table 3.3.4-1	3.3.4
Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	Function 3. Reactor Coolant Outlet Temperature - Hot Leg Function 9. Steam Generator Pressure	
Function 5.B Steam Generator Water Level--High-High	Function 10. Steam Generator Water Level	
Function 6.B Steam Generator Water Level--Low-Low	Function 10. Steam Generator Water Level	
Function 10.C interlock P-12 Tavg--Low-Low	Function 3. Reactor Coolant Temperature - Hot Leg	
Function 10.D interlock P-14 Steam Generator Water Level--High-High	Function 10. Steam Generator Water Level	

3.3 INSTRUMENTATION

3.3.3 Accident Monitoring Instrumentation

LCO 3.3.3 The accident monitoring instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE.

-------------------------NOTE-------------------------

The Provisions of LCO 3.0.4 are not applicable.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels for one or more functions listed in Table 3.3.3-1 inoperable.	A.1 Enter the Condition(s) referenced in Table 3.3.3-1 for each inoperable function.	Immediately

(continued)

Table 3.3.3-1 (Page 1 of 2)

Accident Monitoring Instrumentation

	FUNCTION	REQUIRED NUMBER OF CHANNELS	CONDITION
1.	Pressurizer Water Level	2	B,C
2.	Reactor Coolant Pressure (Wide Range)	2	B,C
3.	Reactor Coolant Outlet Temperature - Thot (Wide Range)	2	B,C
4.	Reactor Coolant Inlet Temperature - Tcold (Wide Range)	2	B,C
5.	Reactor Vessel Water Level	2	B,C
6.	Reactor Coolant System Subcooling Margin Monitor	2	B,D
7.	In Core Thermocouples	4/core quadrant	E
8.	PORV Position Indicator	2/valve	F,G
9.	PORV Block Valve Position Indicator	2/valve	F,G
10.	Safety Valve Position Indicator	2/valve	F,G
11.	Containment Pressure	2	B,C
12.	Containment Sump Water Level	2	B,C
13.	Refueling Water Storage Tank Water Level	2	B,C
14.	Steam Generator Water Level (Narrow Range)	1/steam generator	H
15.	Steam Generator Water Level (Wide Range)	1/steam generator	H
16.	Steam Line Pressure	2/steam generator	I,J

(continued)

Table 3.3.3-1 (Page 2 of 2)

Accident Monitoring Instrumentation

	FUNCTION	REQUIRED NUMBER OF CHANNELS	CONDITION
17.	Auxiliary Feedwater Flow Rate	2/steam generator	I,J
18.	Essential Raw Cooling Water Flow	2	B,C
19.	Containment Atmosphere - High Range Monitor	1/upper containment <u>AND</u> 1/lower containment	K
20.	Shield Building Vent - High Range Noble Gas Monitor	1	L
21.	Steam Line Relief - Noble Gas Monitor	1	L
22.	Condenser Vacuum Exhaust - High Range Noble Gas Monitor	1	L

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required channel inoperable.	B.1 Restore channel to OPERABLE status.	7 days
C. Two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	48 hours
D. Two required channels inoperable.	D.1 Verify RCS subcooling margin within limits.	once per 12 hours
	<u>AND</u>	
	D.2 Restore one channel to OPERABLE status.	48 hours
E. Less than 4 thermocouples per quadrant OPERABLE.	E.1 Restore 4 thermocouples per quadrant to OPERABLE status.	48 hours
F. One required channel per valve inoperable.	F.1 Restore channel to OPERABLE status.	-----NOTE------ Completion Time is on a per valve basis. --------------- 7 days
G. Two required channels per valve inoperable.	G.1 Restore one channel to OPERABLE status.	-----NOTE------ Completion Time is on a per valve basis. --------------- 48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. No required channels per steam generator OPERABLE.	H.1 Restore one channel to OPERABLE status.	-----NOTE------ Completion Time is on a per steam generator basis. -------------- 48 hours
I. One required channel per steam generator inoperable.	I.1 Restore channel to OPERABLE status.	-----NOTE------ Completion Time is on a per steam generator basis. --------------- 7 days
J. Two required channels per steam generator inoperable.	J.1 Restore one channel to OPERABLE status.	-----NOTE------ Completion Time is on a per steam generator basis. -------------- 48 hours
K. Less than 2 containment radiation level channels OPERABLE.	K.1 Establish an alternate method of monitoring.	72 hours
L. One required radiation level channel inoperable.	L.1 Establish an alternate method of monitoring.	72 hours
M. Required Actions and associated Completion Times not met.	M.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required accident monitoring instrumentation channel.	31 days
SR 3.3.3.2	------------------NOTE-------------------- The containment radiation level instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one-point calibration check of the detector below 10 R/hr with a gamma source. -- Perform CHANNEL CALIBRATION for each required accident monitoring instrumentation channel.	18 months

CROSS-REFERENCES

TITLE		NUMBER
Administrative Controls		5.10.7
<u>Accident Monitoring Instrumentation Table 3.3.3-1</u> Function 1. Pressurizer Water Level Function 3. Reactor Coolant Outlet Temperature - Thot (Wide Range)	<u>Reactor Trip System Table 3.3.1-1</u> Function 10. Pressurizer Water Level--High Function 6 Overtemperature ΔT Function 7. Overpower ΔT	3.3.1

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
<u>Accident Monitoring Instrumentation Table 3.3.3-1</u>	<u>Reactor Trip System Table 3.3.1-1</u>	3.3.1
Function 4. Reactor Coolant Inlet Temperature - Tcold (Wide Range)	Function 6. Overtemperature ΔT Function 7. Overpower ΔT	
Function 14. Steam Generator Water Level (Narrow Range)	Function 14. Steam Generator Water Level--Low-Low Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
Function 16. Steam Line Pressure	Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	
<u>Accident Monitoring Instrumentation Table 3.3.3-1</u>	<u>ESFAS Table 3.3.2-1</u>	3.3.2
Function 3. Reactor Coolant Outlet Temperature - Thot	Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 10.C Interlock 9-12 Tavg--Low-Low	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Accident Monitoring Instrumentation Table 3.3.3-1	ESFAS Table 3.3.2-1	3.3.2
Function 4. Reactor Coolant Inlet Temperature - Tcold	Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 10.C Interlock P-12 Tavg--Low-Low	
Function 11. Containment Pressure	Function 1.C Containment Pressure--High Function 2.C Containment Pressure-High-High Function 3.B.3 Containment Pressure--High-High Function 4.B Containment Pressure--High-High	
Function 12. Containment Sump Water Level	Function 7.B Refueling Water Storage Tank (RWST) Level--Low-Low Coincident With Containment Sump Level--High AND Coincident With Safety Injection	
Function 13. Refueling Water Storage Tank Water Level	Function 7.B Refueling Water Storage Tank (RWST) Level--Low-Low Coincident With Containment Sump Level--High AND Coincident With Safety Injection	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Accident Monitoring Instrumentation Table 3.3.3-1	ESFAS Table 3.3.2-1	3.3.2
Function 14. Steam Generator Water Level (Narrow Range)	Function 5.B Steam Generator Water Level--High-High Function 6.B Steam Generator Water Level--Low-Low Function 10.D Interlock P-14 Steam Generator Water Level--High-High	
Function 16. Steam Line Pressure	Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	
Accident Monitoring Instrmentation Table 3.3.3-1	Remote Shutdown System Table 3.3.4-1	3.3.4
Function 1. Pressurizer Water Level	Function 1. Pressurizer Water Level	
Function 3. Reactor Coolant Outlet Temperature - Thot (Wide Range)	Function 3. Reactor Coolant Temperature - Hot Leg	
Function 11. Containment Pressure	Function 7. Containment Pressure	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Accident Monitoring Instrumentation Table 3.3.3-1	Remote Shutdown System Table 3.3.4-1	3.3.4
Function 14. Steam Generator Water Level (Narrow Range)	Function 10. Steam Generator Water Level	
Function 16. Steam Line Pressure	Function 9. Steam Generator Pressure	
Function 17. Auxiliary Feedwater Flow Rate	Function 11. Auxiliary Feedwater Flow Rate	

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System shall be OPERABLE with:

a. Instrumentation channels shown in Table 3.3.4-1, and

b. Transfer switches and controls of system components required for safety-grade:

1. Reactivity control,
2. Reactor Coolant System pressure control,
3. Decay heat removal via auxiliary feedwater flow and steam generator atmospheric dump valve flow,
4. Decay heat removal via the Residual Heat Removal System,
5. Reactor Coolant System inventory control via charging flow, and
6. Safety support systems required for the above functions

--------------------------NOTE---------------------------

The provisions of LCO 3.0.4 are not applicable.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels for one or more functions listed in Table 3.3.4-1 inoperable.	A.1 Enter the Condition(s) referenced in Table 3.3.4.1 for each inoperable function.	Immediately

(continued)

Table 3.3.4-1 (Page 1 of 1)

Remote Shutdown System Instrumentation

	FUNCTION	LOCATION	REQUIRED NUMBER OF CHANNELS	CONDITION
1.	Pressurizer Water Level	ACRP	2	B
2.	Pressurizer Pressure (Wide Range)	ACRP	1	B
3.	Reactor Coolant Temperature--Hot Leg	ACRP	1/loop	C
4.	Source Range Neutron Flux	ACRP	1	B
5.	Reactor Trip Breakers Open/Closed Indication	[local]	1/breaker	D
6.	Control Rod Bottom Bistable	ACRP	1/rod	E
7.	Containment Pressure	ACRP	1	B
8.	Pressurizer Relief Tank Pressure	ACRP	1	B
9.	Steam Generator Pressure	ACRP	1/steam generator	F
10.	Steam Generator Water Level	ACRP	2/steam generator	F
11.	Auxiliary Feedwater Flow Rate	ACRP	1/steam generator	F
12.	RHR Flow Rate	ACRP	1/train	G
13.	RHR Temperature	ACRP	1/train	G

ACRP - Auxiliary Control Room Panel 1-L-10

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more required channels inoperable.	B.1 Restore channels to OPERABLE status.	30 days
C. One or more required channels per loop inoperable.	C.1 Restore channels to OPERABLE status.	30 days
D. One or more reactor trip breaker open/ closed indication circuits inoperable.	D.1 Restore circuits to OPERABLE status.	------NOTE----- Completion Time is on a per reactor trip breaker basis --------------- 30 days
E. One or more required channels per rod inoperable, as required in Table 3.3.4-1	E.1 Restore channels to OPERABLE status.	30 days
F. One or more required channels per steam generator inoperable.	F.1 Restore channels to OPERABLE status.	------NOTE----- Completion Time is on a per steam generator basis --------------- 30 days
G. One or more required channels per train inoperable.	G.1 Restore channels to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. One or more required transfer switches or control circuits inoperable.	H.1 Restore transfer switches and control circuits to OPERABLE status.	------NOTE----- Completion Time is on a per transfer switch or per control circuit basis --------------- 30 days
I. Required Actions and associated Completion Times not met.	I.1 Be in MODE 3.	6 hours
	AND	
	I.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL CHECK for each required Remote Shutdown System instrumentation channel.	31 days
SR 3.3.4.2	Verify each required transfer switch and control circuit performs its intended function.	18 months
SR 3.3.4.3	-------------------NOTE------------------ The following are not applicable for this surveillance: a. Nuclear Instrumentation System source range channels, and b. Reactor trip breaker open/closed indication. ---	
	Perform CHANNEL CALIBRATION for each required Remote Shutdown System instrumentation channel.	18 months

CROSS-REFERENCES

TITLE		NUMBER
Administrative Controls		5.10.7
<u>Remote Shutdown Table 3.3.4-1</u>	<u>Reactor Trip System Table 3.3.1-1</u>	3.3.1
Function 1. Pressurizer Water Level	Function 10. Pressurizer Water Level--High	
Function 3. Reactor Coolant Temperature - Hot Leg	Function 6. Overtemperature ΔT Function 7. Overpower ΔT	
Function 4. Source Range Neutron Flux	Function 5. Source Range Neutron Flux	
Function 5. Reactor Trip Breaker Open/Closed Indication	Function 19. Reactor Trip Breakers	
Function 9. Steam Generator Pressure	Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater FLow Mismatch	
Function 10. Steam Generator Water Level	Function 14. Steam Generator Water Level--Low-Low Function 15. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Remote Shutdown Table 3.3.4-1	ESFAS Table 3.3.2-1	3.3.2
Function 3. Reactor Coolant Outlet Temperature - Thot	Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 4.C High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low Function 10.C Interlock P-12 Tavg--Low-Low Interlock P-14	
Function 5. Reactor Trip Breaker Open/Closed Indication	Function 10.A Interlock P-4 Reactor Trip	
Function 7. Containment Pressure	Function 1.C Containment Pressure--High Function 2.C Containment Pressure--High-High Function 3.B.3 Containment Pressure--High-High Function 4.B Containment Pressure--High-High	
Function 9. Steam Generator Pressure	Function 1.F High Steam Flow in Two Steam Lines Coincident With Tavg--Low-Low OR Coincident With Steam Line Pressure--Low	

(continued)

CROSS-REFERENCES (continued)

TITLE		NUMBER
Remote Shutdown Table 3.3.4-1 Function 10. Steam Generator Water Level	ESFAS Table 3.3.2-1 Function 5.B Steam Generator Water Level--High-High Function 6.B Steam Generator Water Level--Low-Low Function 10.D Interlock P-14 Steam Generator Water Level--High-High	3.3.2
Remote Shutdown Table 3.3.4-1 Function 1. Pressurizer Water Level Function 3. Reactor Coolant Temperature - Hot Leg Function 7. Containment Pressure Function 9. Steam Generator Pressure Function 10. Steam Generator Water Level Function 11. Auxiliary Feedwater Flow Rate	Accident Monitoring Table 3.3.3-1 Function 1. Pressurizer Water Level Function 3. Reactor Coolant Outlet Temperature - Thot Function 11. Containment Pressure Function 16. Steam Line Pressure Function 14. Steam Generator Water Level (Narrow Range) Function 17. Auxiliary Feedwater Flow Rate	3.3.3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow 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 provided below:

a. Pressurizer pressure $\geq$ [2204] psig,

b. RCS average temperature $\leq$ [593]°F, and

c. RCS total flow rate $\geq$ [390,000] gpm, equivalent to [ ]% indicated flow.

APPLICABILITY: MODE 1.

------------------------------NOTE------------------------------
Pressurizer pressure limit does not apply during:

a. A THERMAL POWER ramp in excess of [5]% of RATED THERMAL POWER (RTP) per minute, or

b. A THERMAL POWER step in excess of [10]% of RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS DNB parameter outside limit.	A.1 Restore RCS DNB parameter 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 ≥ [2204] psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature ≤ [593]°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate ≥ [ ]%.	12 hours
SR 3.4.1.4	-------------------NOTE------------------- 1. SR 3.0.4 is not applicable. 2. Calibrate the required instrumentation within [14] days prior to performing this Surveillance. -- Verify, by precision heat balance, that RCS total flow rate is within limit.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Reactor Core Safety Limits	2.1
RCS Loops - Modes 1 and 2	3.4.4
Nuclear Enthalpy Rise Hot Channel Factor	3.2.2

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 $\geq$ [551]°F.

APPLICABILITY: MODE 1 with T_{avg} in one or more RCS loop average temperatures < [561]°F,
MODE 2 with T_{avg} in one or more RCS loop average temperatures < [561]°F, $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTIONS	COMPLETION TIME
A. T_{avg} in one or more RCS loops outside limit.	A.1 Restore T_{avg} to within limit.	15 minutes
	OR	
	A.2 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify each RCS loop T_{avg} within limit.	Within 15 minutes prior to achieving criticality AND -----NOTE----- Only required if T_{avg} - T_{ref} deviation alarm not reset -------------- 30 minutes

CROSS-REFERENCES

TITLE	NUMBER
Mode 2 Physics Tests Exceptions	3.1.10

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3

a. The combination of RCS pressure, except pressurizer pressure, and RCS temperature shall be maintained within the limits provided in the RCS PRESSURE AND TEMPERATURE LIMITS REPORT, and

b. The RCS heatup and cooldown rates shall be maintained within the limits provided in the PRESSURE AND TEMPERATURE LIMITS REPORT.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE-------- All Required Actions must be completed whenever this condition is entered. ------------------- RCS pressure and temperature outside limits. <u>OR</u> RCS heatup or cooldown rate exceeded.	A.1 Restore RCS pressure and temperature to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation by performing an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS.	30 minutes 6 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 with RCS pressure < 500 psig.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the combination of RCS pressure and RCS temperature and the heatup and cooldown rates are within limits.	-----NOTE----- Only required during heatup, cooldown, and in-service leak and hydrostatic testing -------------- 30 minutes

CROSS-REFERENCES

TITLE	NUMBER
Reactor Core Safety Limits	2.1.1
Reactor Coolant System Pressure Safety Limit	2.1.2
RCS Pressure, Temperature, and Flow DNB Limits	3.4.1
RCS Minimum Temperature For Criticality	3.4.2
Cold Overpressure Mitigation System	3.4.16
RCS Pressure/Temperature Limits Report	5.10.9

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - Modes 1 and 2

LCO 3.4.4 Each RCS loop shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS loops not in operation.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop operating and circulating reactor coolant.	12 hours
SR 3.4.4.2	Demonstrate each steam generator OPERABLE in accordance with the Inservice Inspection and Testing Program and Specification 5.9.11.	In accordance with the Inservice Inspection and Testing Program and Specification 5.9.11

CROSS-REFERENCES

TITLE	NUMBER
RCS Loops - Test Exceptions	3.4.17
Steam Generator Tube Inspection Program	5.9.11

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - Mode 3

LCO 3.4.5 At least [two] RCS loops shall be OPERABLE with:

a. At least [two] RCS loops in operation when the reactor trip breakers are closed, or

b. At least one RCS loop in operation when the reactor trip breakers are open.

---------------------------NOTE---------------------------

All reactor coolant pumps may be de-energized for up to 1 hour 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: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status. OR A.2 Be in MODE 4.	72 hours 84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Only one RCS loop in operation and reactor trip breakers closed.	B.1 Restore one RCS loop to operation.	1 hour
	OR	
	B.2 Open reactor trip breakers.	1 hour
C. No RCS loop in operation.	C.1 Open reactor trip breakers.	Immediately
	AND	
	C.2 Suspend all operations involving a reduction in RCS boron concentration.	Immediately
	AND	
	C.3 Initiate action to restore one RCS loop to operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify steam generator secondary-side water level ≥ [10]% (wide range indication) for required OPERABLE RCS loops.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify required number of RCS loops operating and circulating reactor coolant.	12 hours
SR 3.4.5.3	Verify, by correct breaker alignment and indicated power availability, that the required RCPs, which are not in operation, are OPERABLE.	7 days
SR 3.4.5.4	Perform SR 3.4.4.2 (Steam Generator OPERABILITY).	In accordance with SR 3.4.4.2

CROSS-REFERENCES - None.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - Mode 4

LCO 3.4.6 At least two loops consisting of any combination of RCS loops and Residual Heat Removal (RHR) loops, shall be OPERABLE, and at least one loop shall be in operation.

---------------------------NOTES---------------------------

1. All Reactor Coolant Pumps (RCPs) and RHR pumps may be de-energized for up to 1 hour provided:

   a. No operations are permitted that would cause dilution 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 one or more RCS cold leg temperatures $\leq$ [310]°F unless the secondary-side water temperature of each Steam Generator (SG) is < [50]°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Only one RHR loop OPERABLE. AND No RCS loops OPERABLE.	A.1 Restore a second loop to OPERABLE status. OR A.2 Be in MODE 5.	1 hour 25 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Only one RCS loop OPERABLE. AND No RHR loops OPERABLE.	B.1 Initiate action to return a second loop to OPERABLE status.	Immediately
C. No loop in operation.	C.1 Suspend all operations involving a reduction in RCS boron concentration. AND C.2 Initiate action to restore one RHR or RCS loop to operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify SG secondary-side water levels ≥ [10]% (wide range indication) for required OPERABLE RCS loops.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.6.2	Verify at least one RHR or RCS loop operating and circulating reactor coolant.	12 hours
SR 3.4.6.3	Verify, by correct breaker alignment and indicated power availability, that the required pump, which is not in operation, is OPERABLE.	7 days
SR 3.4.6.4	Perform SR 3.4.4.2	In accordance with SR 3.4.4.2

CROSS-REFERENCES

TITLE	NUMBER
ECCS Trains - Shutdown (T_{avg} < 350°F)	3.5.3
Cold Overpressure Mitigation System	3.4.16

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - Mode 5, Loops Filled

LCO 3.4.7 At least 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 [two] Steam Generators (SGs) shall be ≥ [10]% (wide range indication).

---------------------------NOTES---------------------------

1. The RHR pump of the RHR loop in operation may be de-energized for up to 1 hour provided:

    a. No operations are permitted that would cause dilution of the RCS boron concentration, and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One RHR loop may be inoperable for up to 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 ≤ [310]°F unless the secondary-side water temperature of each SG is < [50]°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Less than [two] SGs level within limit.	A.1 Initiate action to restore RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG levels to within limit.	Immediately Immediately
B. No RHR loop OPERABLE or in operation.	B.1 Suspend all operations involving a reduction in RCS boron concentration. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify secondary-side water levels is ≥ [10]% (wide range indication) in at least [two] steam generators.	-----NOTE----- Only required if only one RHR loop is OPERABLE ------------- 12 hours
SR 3.4.7.2	Verify at least one RHR loop operating and circulating reactor coolant.	12 hours
SR 3.4.7.3	Verify, by correct breaker alignment and indicated power availability, that the required RHR loop, which is not in operation, is OPERABLE.	-----NOTE----- Only required if secondary-side water level is ≤ [10]% (wide range indication) in more than 2 SGs --------------- 7 days

CROSS-REFERENCES

TITLE	NUMBER
RCS Loops - Mode 5, Loops Not Filled	3.4.8
Cold Overpressure Mitigation System	3.4.16

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 at least one RHR loop shall be in operation.

---------------------------NOTES---------------------------

1. The RHR pump of the RHR loop in operation may be de-energized for up to 1 hour provided:

    a. No operations are permitted that would cause dilution of the RCS boron concentration, and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One RHR loop may be inoperable for up to 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. Only one RHR loop inoperable.	A.1 Initiate action to restore two RHR loops to OPERABLE status.	Immediately
B. No RHR loop OPERABLE or in operation.	B.1 Suspend all operations involving a 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 at least one RHR loop operating and circulating reactor coolant.	12 hours
SR 3.4.8.2	Verify, by correct breaker alignment and indicated power availability, that the required RHR loop, which is not in operation, is OPERABLE.	7 days

CROSS-REFERENCES

TITLE	NUMBER
RCS Loops - Mode 5, Loops Filled	3.4.7
Cold Overpressure Mitigation System	3.4.16

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 volume outside limit.	A.1 Restore pressurizer water volume to within limit.	1 hour
B. One required group of pressurizer heaters inoperable.	B.1 Restore at least 2 groups of pressurizer heaters to OPERABLE status.	72 hours
C. Required Actions and associated Completion Time of Condition A not met.	C.1 Be in MODE 3 with reactor trip breakers open. AND C.2 Be in MODE 4.	6 hours 12 hours
D. Required Action and associated Completion Time of Condition B not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water volume ≤ [1656] cubic feet, equivalent to ≤ [92]% of span.	12 hours
SR 3.4.9.2	Verify capacity of 2 required groups of pressurizer heaters with capacity of ≥ [150] kw.	92 days
SR 3.4.9.3	Demonstrate emergency power supply for required groups of pressurizer heaters is OPERABLE by transferring from the normal power supply to the emergency power supply and energizing the heaters.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Reactor Trip System Instrumentation Function 10	3.3.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Each pressurizer code safety valve shall be OPERABLE with lift settings ≥ [2460] and ≤ [2510] psig.

---NOTES---

1. The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating RCS temperature and pressure.

2. The provisions of LCO 3.0.4 and SR 3.0.4 are not applicable for entry into MODE 3 and may be suspended for up to 18 hours after entry into MODE 3 for the purpose of setting the pressurizer code safety valves (one at a time) under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with one or more RCS Cold Leg temperatures > [310]°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer code safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with all RCS Cold Leg temperatures ≤ [310]°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.10.1	Demonstrate each pressurizer code safety valve OPERABLE in accordance with the Inservice Inspection and Testing Program.	In accordance with the Inservice Inspection and Testing Program

CROSS-REFERENCES

TITLE	NUMBER
Inservice Inspection and Testing Program	5.9.14

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.

---NOTE---
The provisions of LCO 3.0.4 are not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable due to excessive leakage.	A.1 Restore PORVs to OPERABLE status. OR A.2 Close associated block valves.	1 hour 1 hour
B. One PORV inoperable due to causes other than seat excessive leakage.	B.1 Restore PORV to OPERABLE status. OR B.2.1 Close associated block valve AND B 2.2 Remove power from associated block valve. AND B.2.3 Restore PORV to OPERABLE status. (continued)	1 hour 1 hour 73 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	<u>AND</u>	
	B.2.4 Restore power to associated block valve.	73 hours
	<u>AND</u>	
	B.2.5 Open associated block valve.	73 hours
C. One block valve inoperable.	C.1 Restore block valve to OPERABLE status.	1 hour
	<u>OR</u>	
	C.2.1 Close block valve.	1 hour
	<u>AND</u>	
	C.2.2 Remove power from block valve.	
	<u>AND</u>	
	C.2.3 Restore block valve to OPERABLE status.	73 hours
	<u>OR</u>	
	C.3.1 Deactivate associated PORV by removing power from its solenoid valve.	1 hour
	<u>AND</u>	
	C.3.2 Restore block valve and PORV to OPERABLE status.	73 hours
D. Required Actions of Conditions A, B, or C <u>NOT</u> met within required Completion Times.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. More than one PORV inoperable due to causes other than excessive leakage.	E.1 Restore PORVs to OPERABLE status.	1 hour
	<u>OR</u>	
	E.2.1 Close associated block valves.	1 hour
	<u>AND</u>	
	E.2.2 Remove power from associated block valve.	
	<u>AND</u>	
	E.2.3 Be in MODE 3.	7 hours
	<u>AND</u>	
	E.2.4 Be in MODE 4.	13 hours
F. More than one block valve inoperable.	F.1 Restore block valves to OPERABLE status.	1 hour
	<u>OR</u>	
	F.2.1.1 Close and remove power from inoperable block valves.	1 hour
	<u>OR</u>	
	F.2.1.2 Close and deactivate associated PORVs by removing power from their solenoid valves.	1 hour
	<u>AND</u>	
	F.2.2 Be in MODE 3.	7 hours
	<u>AND</u>	
	F.2.3 Be in MODE 4.	13 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Operate each block valve through one complete cycle of full travel (unless closed in accordance with the Required Actions of this specification).	92 days
SR 3.4.11.2	Perform CHANNEL CALIBRATION for each PORV.	18 months
SR 3.4.11.3	Operate each PORV through one complete cycle of full travel.	18 months
SR 3.4.11.4	Demonstrate emergency power supply for PORVs and block valves is OPERABLE by transferring from normal power supply to the emergency power supply and by operating the valves through a complete cycle of full travel.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Cold Overpressure Mitigation System	3.4.16

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 RCS Operational Leakage

LCO 3.4.12 The RCS operational leakage shall be limited to:

a. No pressure boundary LEAKAGE,

b. One gpm unidentified LEAKAGE,

c. Ten gpm identified LEAKAGE from the RCS,

d. One gpm total primary-to-secondary leakage through all steam generators, and

e. [500] gallons per day primary-to-secondary leakage through any one steam generator.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists.	A.1 Be in MODE 3. AND A.2 Be in Mode 5.	6 hours 36 hours
B. RCS leakage outside limits for reasons other than pressure boundary LEAKAGE.	B.1 Reduce leakage to within limit.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Actions and associated Completion Times of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
	------------------NOTE------------------- The provisons of SR 3.0.4 are not applicable for entry into and operation in MODES 3 or 4. ---	
SR 3.4.12.1	Perform a RCS water inventory balance.	72 hours during steady state operation.
SR 3.4.12.2	Verify primary-to-secondary leakage limited to 1 gpm through all steam generators and [500] gallons per day through any one steam generator.	72 hours during steady state operation.

CROSS-REFERENCES

TITLE	NUMBER
RCS Pressure Isolation Valve Leakage	3.4.13
RCS Leakage Detection Instrumentation	3.4.14

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.13 The leakage from each RCS Pressure Isolation Valve (PIV) shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a RCS pressure ≥ [2215] and ≤ [2255] psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage from one or more RCS PIVs outside limit.	A.1 Restore RCS PIV leakage to within limit.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------NOTE------------------ The provisions of SR 3.0.4 are not applicable for entry into MODE 3 and 4, for the purpose of testing the isolation check valves --- Verify leakage of each RCS PIV within limits.	18 months <u>AND</u> Prior to entry into MODE 2, whenever the unit has been in MODE 5 for > 72 hours if testing has not been performed in the previous 9 months <u>AND</u> Within 24 hours following valve actuation due to automatic or manual action, or flow through the valve <u>AND</u> In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.13.2	Verify Residual Heat Removal (RHR) System auto-closure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal ≥ [380] psig.	18 months

CROSS-REFERENCES

TITLE	NUMBER
RCS Loops - MODE 4	3.4.6
RCS Operational Leakage	3.4.12
Accumulators	3.5.1
ECCS Trains - Operating ($T_{avg} \geq 350°F$)	3.5.2
Containment Isolation Valves	3.6.6

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Leakage Detection Instrumentation

LCO 3.4.14 The RCS Leakage Detection Instrumentation shall be OPERABLE with:

a. One containment atmosphere (gaseous or particulate) radioactivity monitor, and

b. One containment floor and equipment drain sump level monitor.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment atmosphere radioactivity monitor inoperable. AND All other required monitors OPERABLE.	A.1 Take and analyze grab samples of the containment atmosphere. AND A.2 Restore containment atmosphere radioactivity monitor to OPERABLE status.	Once per 24 hours 30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required containment sump level monitor inoperable. AND All other required monitors OPERABLE.	B.1 Perform SR 3.4.12.1. AND B.2.1 Restore containment sump level monitor to OPERABLE status.	Once per 12 hours 30 days
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	Perform CHANNEL CHECK of required atmosphere gaseous and particulate radioactivity monitoring instrumentation.	12 hours
SR 3.3.14.2	Perform ANALOG CHANNEL OPERATIONAL TEST for required containment atmosphere radioactivity monitor.	31 days
SR 3.4.14.3	Perform CHANNEL CALIBRATION of containment atmosphere radioactivity monitor and containment floor and equipment drain sump level monitor.	18 months

CROSS-REFERENCES

Title	Number
RCS Operational Leakage	3.4.12

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Specific Activity

LCO 3.4.15 The specific activity of the primary coolant shall be limited to:

a. A gross specific activity of $\leq 100/\bar{E}$ μCi/g,

b. A DOSE EQUIVALENT I-131 specific activity of ≤ 1.0 μCi/g, and

c. A DOSE EQUIVALENT I-131 specific activity within the acceptable region of Figure 3.4.15-1.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature $\geq$ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross specific activity of the primary coolant outside limit.	A.1 Perform SR 3.4.15.3	Once per 4 hours
	AND	
	A.2 Be in MODE 3 with RCS average temperature < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Specific activity of the primary coolant > 1.0 μCi/g DOSE EQUIVALENT I-131 specific activity but within the acceptable region of Figure 3.4.15-1.	B.1 Perform SR 3.4.15.3 __AND__ B.2 Restore specific activity to within limit.	Once per 4 hours 48 hours
C. Specific activity of the primary coolant in the unacceptable region of Figure 3.4.15-1. __OR__ Required Actions and associated Completion Time of Condition B not met.	C.1 Perform SR 3.4.15.3. __AND__ C.2 Be in MODE 3 with RCS average temperature < 500°F.	Once per 4 hours 6 hours

SURVEILLANCE REQUIREMENTS

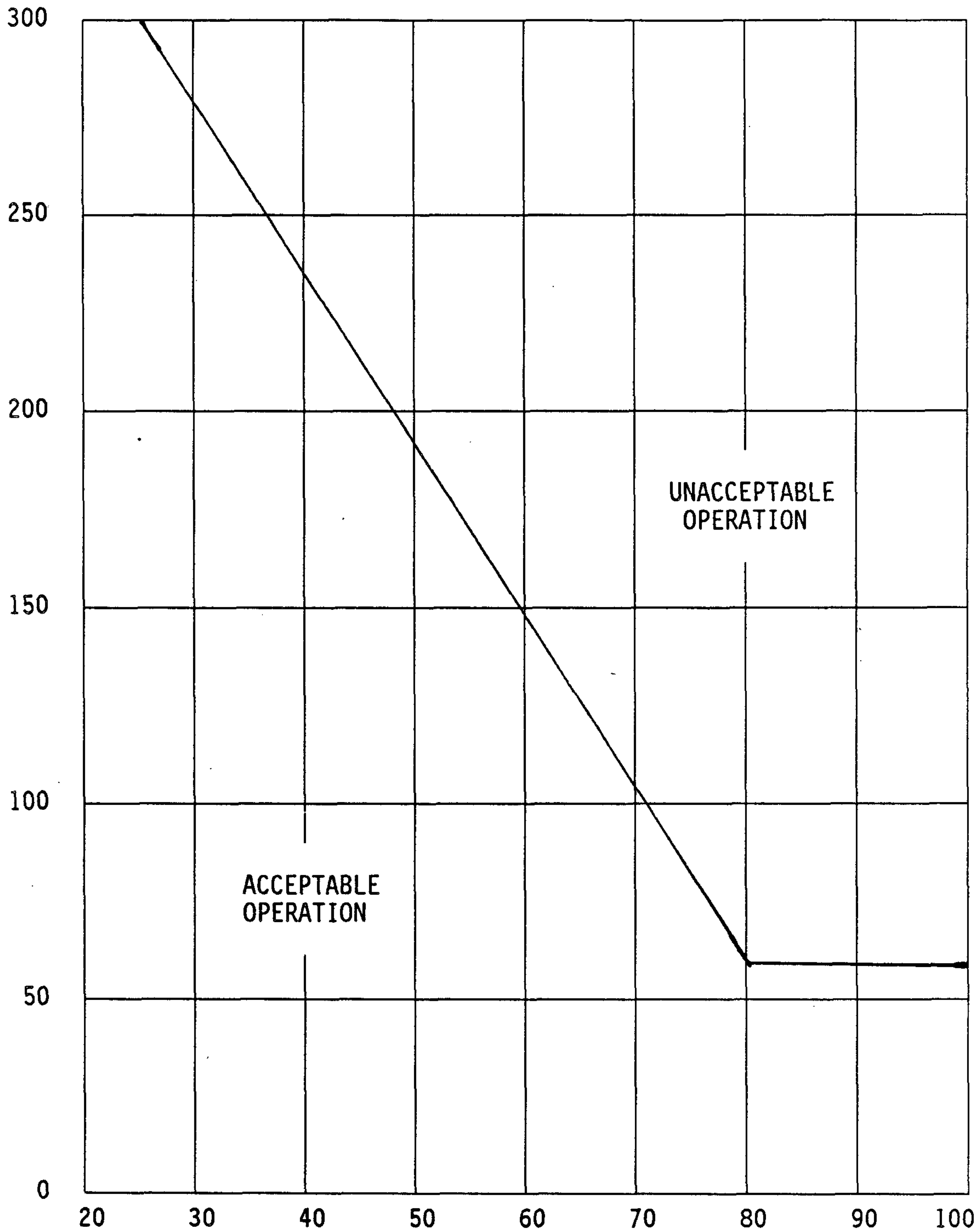
SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform gamma isotopic analysis inclusive of DOSE EQUIVALENT I-131 determination.	72 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.15.2	-------------------NOTE------------------ Sample after a minimum of 2 EFPD or 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. -- Determine $\bar{E}$ by radiochemical analysis.	-----NOTE----- Only required in MODE 1 ------------- 184 days
SR 3.4.15.3	Verify primary coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 1.0 μCi/g by performing a short isotopic analysis using I-131, I-133, and I-135.	From 2 to 6 hours after a THERMAL POWER change of $\geq$ 15% RATED THERMAL POWER within a 1-hour period

CROSS-REFERENCES - None.



Percent of RATED THERMAL POWER

Figure 3.4.15-1 (Page 1 of 1)

Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit
Versus Percent of RATED THERMAL POWER

Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit (μCi/g)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Cold Overpressure Mitigation System (COMS)

LCO 3.4.16 A maximum of [1] [centrifugal charging pump] shall be OPERABLE, and a Cold Overpressure Mitigation System shall be OPERABLE with:

a. Two Power-Operated Relief Valves (PORVs) with nominal lift settings within the limits provided in Figure 3.4.16-1, or

b. The RCS depressurized with an open RCS vent of $\geq$ [3] square inches.

APPLICABILITY: MODE 4 with one or more RCS cold leg temperatures $\leq$ [310]°F,
MODE 5,
MODE 6 with the reactor vessel head on.

---------------------------NOTE---------------------------
The provisions of LCO 3.0.4 are not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required PORV inoperable.	A.1 Restore required PORV to OPERABLE status.	7 days
B. Two PORVs inoperable. OR Required Action and associated Completion Time of Condition A or B not met.	B.1 Depressurize RCS and establish RCS vent of $\geq$ [3] square inches.	8 hours

(continued)

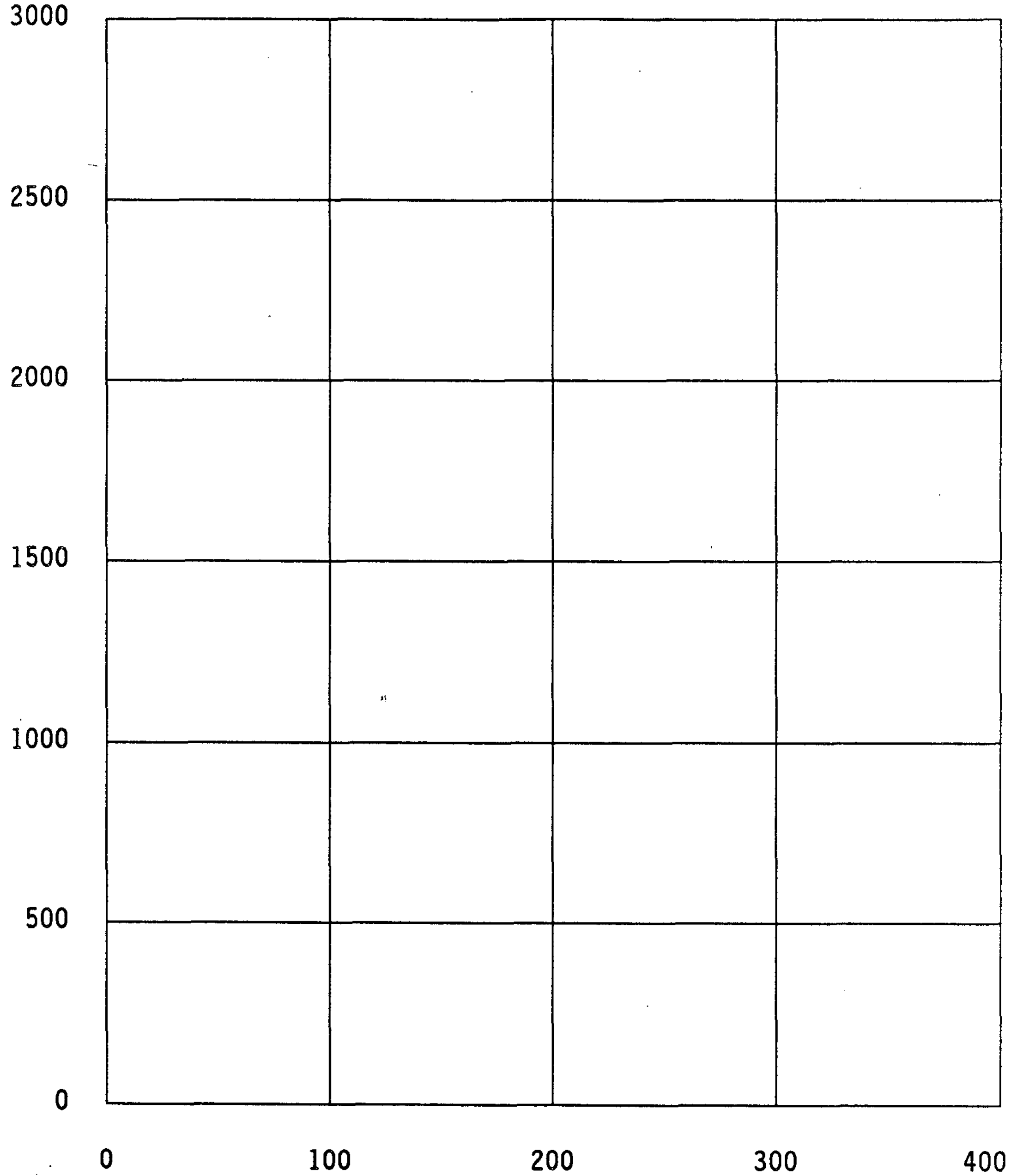
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify a maximum of [1] centrifugal charging pump OPERABLE.	Within 15 minutes of decreasing temperature to ≤ [310]°F AND 12 hours
SR 3.4.16.2	-----------------NOTE-------------------- SR 3.0.4 is not applicable. -- Verify the RHR suction valves are open.	12 hours
SR 3.4.16.3	Verify required RCS vents open: a. For unsecured-open vent valves, and b. For secured-open vent valves.	 12 hours 31 days
SR 3.4.16.4	Verify the block valve open for each required PORV.	72 hours
SR 3.4.16.5	Verify RHR suction isolation valves are open with power to the valve operator removed.	31 days
SR 3.4.16.6	Perform ANALOG CHANNEL OPERATIONAL TEST, excluding valve operation, for each required PORV.	31 days
SR 3.4.16.7	Perform CHANNEL CALIBRATION of each required PORV.	18 months

CROSS-REFERENCES

TITLE	NUMBER
RCS Pressure and Temperature Limits	3.4.3
Pressurizer Power-Operated Relief Valves	3.4.11
ECCS Trains - Shutdown (T_{avg} < 350°F)	3.5.3
Safety Limit Violation	5.7
Special Reports	5.10.7

* TO BE PROVIDED



[Auctioneered Low Measured] RCS Temperature, (°F)

Figure 3.4.16-1 (Page 1 of 1)

PORV Setpoint Versus RCS Temperature

[Maximum Nominal Allowed] PORV Setpoint, (psig)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Loops - Test Exceptions

LCO 3.4.17 The requirements of LCO 3.4.4, RCS Loops - Modes 1 and 2, may be suspended provided:

a. THERMAL POWER does not exceed the P-7 Interlock Setpoint, [10]%, and

b. The reactor trip setpoint of the OPERABLE intermediate and power range channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: During performance of startup and PHYSICS TESTS with THERMAL POWER < the P-7 Interlock setpoint.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER > P-7 Interlock setpoint.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify THERMAL POWER is < P-7 Interlock setpoint.	1 hour
SR 3.4.17.2	Perform ANALOG CHANNEL OPERATIONAL TEST for each Power Range Neutron Flux--Low and Intermediate Range Neutron Flux channel and P-7 Interlock.	Within 12 hours prior to initiation of startup and PHYSICS TESTS

CROSS-REFERENCES

TITLE	NUMBER
Reactor Trip System Instrumentation	3.3.1
RCS Loops - Modes 1 and 2	3.4.4

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure $\geq$ [1000] 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 Actions and associated Completion Times of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Reduce pressurizer pressure to < [1000] psig.	6 hours 12 hours
D. More than one accumulator inoperable.	D.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve fully open.	12 hours
SR 3.5.1.2	Verify contained borated water volume in each accumulator is ≥ [7627] and ≤ [8082] gal.	12 hours
SR 3.5.1.3	Verify nitrogen cover-pressure in each accumulator is ≥[585] and ≤[678] psig.	12 hours
SR 3.5.1.4	Verify boron concentration for each accumulator is ≥[1900] ppm and ≤[2100] ppm.	31 days AND Once within 6 hours after solution volume changes ≥ [75] gallons of the affected tank volume which is not the result of an addition from the RWST.
SR 3.5.1.5	---------------NOTE------------------ Only required when RCS pressure is ≥[1000] psig. ------------------------------------- Verify power removed from each accumulator isolation valve operator.	31 days

CROSS-REFERENCES - None.

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.2 ECCS Trains - Operating ($T_{avg} > 350°F$)

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

------------------------NOTES---------------------------

1. In MODE 3 both Safety Injection pump flow paths may be isolated for a period up to 2 hours to perform Pressure Isolation Valve testing per SR 3.4.13.1.

2. LCO 3.0.4 and SR 3.0.4 are not applicable for entry into MODE 3 for the pump(s) declared inoperable pursuant to LCO 3.4.16, Cold Overpressure Prevention. This exepmtion is allowed for up to 4 hours following entry into MODE 3 or prior to the temperature of one or more of the Reactor Coolant System cold legs exceeding 375°F, whichever comes first.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more components inoperable. AND At least 100% of the safety injection flow equivalent to a single train available.	A.1 Restore components to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Valve Number / Position / Function [FCV-63-1] / [open] / [RHR Supply] [FCV-63-22] / [open] / [SIS Discharge]	12 hours
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 its correct position.	31 days
SR 3.5.2.3	Demonstrate ECCS piping is full of water.	31 days
SR 3.5.2.4	Demonstrate each ECCS pump's developed head at the test flow point is ≥ the required head.	In accordance with the Inservice Inspection and Testing Program
SR 3.5.2.5	Demonstrate each ECCS train automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Demonstrate, for each ECCS throttle valve listed below, each position stop is in its correct position. CCP Discharge Throttle Valves: 63-582, 63-583, 63-584, 63-585 SI Cold Leg Throttle Valves: 63-550, 63-552, 63-554, 63-556 SI Hot Leg Throttle Valves: 63-542, 63-544, 63-546, 63-548	18 months
SR 3.5.2.7	Demonstrate each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.8	Verify by visual inspection, that each ECCS train containment sump suction inlet is not restricted by debris and that the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Special Reports	5.10.7

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.3 ECCS Trains - Shutdown (T_{avg} < 350°F)

LCO 3.5.3 One ECCS train shall be OPERABLE.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR pump, heat exchanger or flow path inoperable.	A.1 Initiate Actions to restore one ECCS train to OPERABLE status.	Immediately
B. Required centrifugal charging pump or flow path from RWST inoperable.	B.1 Restore one ECCS train to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 --------NOTE---------- Only required if at least one RHR loop OPERABLE. ---------------------- Be in MODE 5	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	Perform the following surveillances for all equipment required to be OPERABLE. SR 3.5.2.1 SR 3.5.2.5 SR 3.5.2.2 SR 3.5.2.6 SR 3.5.2.3 SR 3.5.2.7 SR 3.5.2.4 SR 3.5.2.8	In accordance with applicable SRs

CROSS-REFERENCES

TITLE	NUMBER
RCS Loops - Mode 4	3.4.6
Cold Overpressure Mitigation System	3.4.16
Special Reports	5.10.7

3.5 EMERGENCY CORE COOLING SYSTEM (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 borated water volume not within limit.	A.1 Restore RWST water volume to within limit.	1 hour
B. RWST boron concentration or water temperature not within limits.	B.1 Restore RWST to OPERABLE status.	8 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature ≥ [60] °F and ≤ [105] °F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume ≥ [370,000] gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ [2000]ppm and ≤ [2100]ppm.	7 days

CROSS-REFERENCES

TITLE	NUMBER
ECCS Trains - Operating (Tavg ≥ 350°F)	3.5.2
ECCS Trains - Shutdown (T_{avg} < 350°F)	3.5.3
Containment Spray System	3.6.3

3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be within limits.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------NOTE------------- The RCP seal injection flow limit is only applicable when the conditions specified in SR 3.5.5.1 are met. ----------------------------- A. Seal injection flow not within limit.	A.1 Reduce flow to within limit. AND A.2 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump discharge header pressure ≥ [2430] psig and the [pressurizer level] control valve full open and RCS pressure is ≥ [2215] and ≤ [2250] psig.	1 hour 4 hours
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	------------------NOTE------------------ SR 3.0.4 is not applicable for entry into MODE 3. This exemption is allowed for up to 4 hours after the reactor coolant system pressure stablizes at $\geq$[2215] psig and $\leq$[2250] psig. --	
	Verify manual seal injection throttle valves adjusted to give a flow $\leq$ [40] gpm with centrifugal charging pump discharge header pressure $\geq$ [2430] psig, the pressurizer level control valve full open, one centrifugal charging pump running and RCS pressure is $\geq$ [2215] psig and $\leq$ [2250] psig.	31 days

CROSS-REFERENCES - None.

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. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leak rate testing except for containment air lock door seal testing, in accordance with 10 CFR 50, Appendix J as modified by approved exceptions as contained in the Containment Leak Rate Testing Program.	-----NOTE----- SR 3.0.2 is not applicable -------------- In accordance with 10 CFR 50, Appendix J as modified by approved exceptions, as contained in the Containment Leak Rate Testing Program

CROSS REFERENCES - None

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Each containment air lock shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any of the following in one or more containment air locks:	-------------NOTE----------- Entry and exit is permissible to perform repairs of the affected air lock components, perform required surveillances, or to make necessary containment entries. ----------------------------	
One door inoperable.	A.1 Verify an OPERABLE door is closed in each affected air lock.	15 minutes
OR	AND	
The interlock mechanism inoperable.	A.2.1 Restore air lock to OPERABLE status.	24 hours
OR	OR	
One door and interlock mechanism inoperable.	A.2.2.1 Lock an OPERABLE door closed in each affected air lock.	24 hours
	AND	
	A.2.2.2 Verify an OPERABLE door is locked closed in each affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment air locks inoperable for reasons other than Condition A.	------------NOTE------------ If both doors in an airlock have catastrophically failed seals, containment shall be declared inoperable in accordance with LCO 3.6.1. ----------------------------	
	------------NOTE------------ Entry and exit is permissible to perform repairs of the affected air lock components, perform required surveillances, and to make necessary containment entries. ----------------------------	
	B.1 Verify a door is closed in each affected air lock.	15 minutes
	<u>AND</u>	
	B.2 Restore to OPERABLE status.	24 hours
C. Required Actions and associated Completion Times 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 3.6.2.1	-------------------NOTE----------------- SR 3.0.2 is not applicable. -- -------------------NOTE----------------- An inoperable air lock door does not invalidate the previous successful performance of an overall air lock leakage test. -- Perform required air lock leak rate testing in accordance with 10 CFR 50, Appendix J as modified by approved exemptions as contained in the Containment Leak Rate Testing Program. The acceptance criteria for air lock testing are: a. Overall air lock leak rate is $\leq$ [0.05 La] when tested at [Pa]. b. For each door, leak rate is $\leq$ [.01La] when tested at $\geq$ [6] psig.	In accordance with 10 CFR 50, Appendix J as modified by approved exemptions, as contained in the Containment Leak Rate Testing Program.
SR 3.6.2.2	Demonstrate that only one door in each air lock can be opened at a time.	-----NOTE------ Only required if not performed within previous 184 days. --------------- Prior to entry into containment

CROSS-REFERENCES

TITLE	NUMBER
Containment	3.6.1

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Spray System (CSS)

LCO 3.6.3 Two CSS trains shall be OPERABLE

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CSS train inoperable.	A.1 Restore CSS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each CSS manual, power-operated and automatic valve in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position.	31 days
SR 3.6.3.2	Demonstrate each CSS pump's developed head at the flow test is $\geq$ the required developed head.	In accordance with the Inservice Inspection and Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.3	Demonstrate each CSS automatic valve in the flowpath actuates to its correct position on an actual or simulated actuation signal(s).	18 months
SR 3.6.3.4	Demonstrate that each CSS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.3.5	Demonstrate each spray nozzle unobstructed.	5 years

CROSS-REFERENCES

TITLE	NUMBER
Inservice Inspection and Testing Program	5.9.14

3.6 CONTAINMENT SYSTEMS

3.6.4 Air Return Fan System (ARFS)

LCO 3.6.4 Two ARFS trains shall be OPERABLE.

APPLICABILITY MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARFS train inoperable.	A.1 Restore ARFS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Demonstrate each ARFS fan starts, on an actual or simulated Containment Pressure High-High Signal after a delay of ≥[9] and ≤[11] minutes, and operates for ≥ 15 minutes.	18 months
SR 3.6.4.2	Demonstrate with the ARFS fan dampers closed, each ARFS fan motor current is ≥ [54] amps and ≤ [94] amps.	92 days
SR 3.6.4.3	Demonstrate with the ARFS fan not operating, each ARFS fan damper opens when ≤ [150] inch-lbs is applied to the counterweight.	92 days

CROSS-REFERENCES - None

3.6 CONTAINMENT SYSTEMS

3.6.5 Emergency Gas Treatment System (EGTS)

LCO 3.6.5 Two EGTS trains shall be OPERABLE and the annulus pressure shall be $\geq$ [5] inches water gauge vacuum with respect to the penetration room on El. 737.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EGTS train inoperable.	A.1 Restore EGTS train to OPERABLE status.	7 days
---------NOTE----------- Annulus pressure requirement is not applicable during venting operations, required annulus entries, or Auxiliary Building isolations not exceeding 1 hour in duration. ------------------------ B. Annulus pressure not within limits.	B.1 Restore annulus pressure within limits.	8 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Operate each EGTS train for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.5.2	Perform required filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.6.5.3	Demonstrate each EGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.6.5.4	Demonstrate each EGTS automatic damper actuates to its correct position on an actual or simulated actuation signal.	18 months
SR 3.6.5.5	Demonstrate each EGTS train produces a pressure equal to or more negative than [-0.5] inches water gauge in the annulus relative to the Mechanical Equipment Room El. 772 within [ ] minute after a start signal with an inleakage of ≤ [500] cfm.	18 months
SR 3.6.5.6	Verify the annulus pressure ≥ [5] inches water gauge vacuum with respect to penetration room on El. 737.	24 hours

CROSS-REFERENCES

TITLE	NUMBER
Ventilation Filter Testing Program	5.9.13

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Isolation Valves

LCO 3.6.6 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE--------- Not applicable to those penetrations with only one containment isolation valve and a closed system. -------------------- One or more containment isolation valves inoperable.	-----------NOTE----------- Isolation valves may be opened intermittently under administrative control. --------------------------	
	A.1 -------NOTE-------- Not applicable to those penetrations that have only one isolation valve. ------------------- Verify at least one isolation valve is OPERABLE in each affected open penetration.	1 hour
	AND	
	A.2.1 Restore the valve to OPERABLE status.	4 hours
	OR	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2.1 Isolate each affected penetration by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve inside containment with flow through the valve secured. AND A.2.2.2 Verify each affected penetration is isolated.	4 hours Once per 31 days for valves not in containment and annulus AND Prior to entering MODE 4 from MODE 5 but not more often than once per 92 days for valves inside containment or annulus
B. --------NOTE---------- Only applicable to those penetrations with only one containment isolation valve and a closed system. ---------------------- One or more containment isolation valves inoperable.	B.1 Restore the valve to OPERABLE status. OR B.2.1 Isolate each affected penetration by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. (continued)	7 days 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	<u>AND</u> B.2.2 Verify each affected penetration is isolated.	Once per 31 days for valves not in containment or the annulus
C. One or more containment purge valves not within purge valve leakage limits.	C.1 Restore leakage to within limits. <u>OR</u> C.2.1 Isolate each affected penetration by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. <u>AND</u> C.2.2 Perform SR 3.6.6.6	24 hours 24 hours Once per 92 days
D. Required Actions and associated Completion Times not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify the 24-inch containment lower compartment purge supply and/or exhaust isolation valves are physically restricted to ≤ [50°] open.	31 days
SR 3.6.6.2	---NOTE--- 1. Valves and blind flanges in high radiation or hazardous areas may be verified by use of administrative controls. 2. Valves may be opened intermittently under administrative controls. 3. Not required to be met on valves which are open under administrative controls. --- Verify all containment isolation manual valves, deactivated automatic valves and blind flanges which are located outside containment, the annulus, or the north and south valve vault rooms, and required to be closed during accident conditions are closed.	31 days
SR 3.6.6.3	---NOTE--- 1. Valves and blind flanges in high radiation or hazardous areas may be verified by use of administrative controls. 2. Valves may be opened intermittently under administrative controls. 3. Not required to be met on valves which are open under administrative controls. --- Verify all containment isolation manual valves, deactivated automatic valves, and blind flanges which are required to be closed are located inside containment, the annulus, or the north and south valve vault rooms are closed.	Prior to entering MODE 4 from MODE 5 but not more often than once per 92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.4	Demonstrate the isolation time of each power-operated and each automatic containment isolation valve is within limits.	In accordance with Inservice Inspection and Testing Program
SR 3.6.6.5	Demonstrate each automatic containment isolation valve actuates to its isolation position on an actual or simulated actuation signal.	18 months
SR 3.6.6.6	------------------NOTE--------------------- Results shall be evaluated against the acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J as modified by approved exemptions as contained in the CLRT Program. --- Perform additional required leak rate testing on containment purge valves with resilient seals in accordance with the Containment Leak Rate Testing Program.	184 days <u>AND</u> within 92 days after opening the valve

CROSS-REFERENCES

TITLE	NUMBER
Containment	3.6.1
Inservice Inspection and Testing Program	5.9.14

3.6 CONTAINMENT SYSTEMS

3.6.7 Containment Internal Pressure

LCO 3.6.7 Containment pressure shall be $\geq$ [-0.1] psid and $\leq$ [+0.3] psid relative to the annulus.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment internal pressure to within limits.	1 hour
B. Required Action not met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify containment pressure $\geq$[-0.1] psid and $\leq$[+0.3] psid relative to the annulus.	12 hours

CROSS-REFERENCES

TITLE	NUMBER
Containment	3.6.1

3.6 CONTAINMENT SYSTEMS

3.6.8 Containment Air Temperature

LCO 3.6.8 Containment average air temperature shall:

a. Be ≥[85] and ≤[110]°F for the containment upper compartment, and

b. Be ≥[100] and ≤[120]°F for the containment lower compartment.

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

------------------------NOTE------------------------
The lower limit may be reduced to 60°F in MODES 2, 3, and 4 for both the upper and lower compartments.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Verify containment upper compartment average air temperature is ≥[85] and ≤[110]°F.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.8.2	Verify containment lower compartment average air temperature, is ≥[100]°F and ≤[120]°F.	24 hours

CROSS-REFERENCES

TITLE	NUMBER
Containment	3.6.1

3.6 CONTAINMENT SYSTEMS

3.6.9 Ice Bed

LCO 3.6.9 The ice bed shall be OPERABLE:

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ice bed inoperable.	A.1 Restore ice bed to OPERABLE status.	48 hours
B. Required Actions and associated Completion Time not met.	B.1 Be in Mode 3. AND B.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.9.1	Verify maximum ice bed temperature $\leq$ [27]°F.	12 hours
SR 3.6.9.2	Verify by chemical analyses of $\geq$ 9 representative samples of stored ice: a. Boron concentration $\geq$[1800] ppm, and b. pH $\geq$[9.0] and $\leq$[9.5] at 20°C.	9 months

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.9.3	Verify total weight of stored ice is ≥ [1,999,800] lbs by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains ≥ [1029] lbs of ice, and b. Calculating total weight of stored ice, at a 95% confidence level, using all ice basket weights determined in SR 3.6.9.3a.	18 months
SR 3.6.9.4	Verify azimuthal distribution of ice at a 95% confidence level by subdividing weights as determined by SR 3.5.9.3a. a. Group 1 - bays 1 through 8, b. Group 2 - bays 9 through 16, and c. Group 3 - bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be ≥ [1029] lbs.	18 months
SR 3.6.9.5	Verify, by visual inspection, accumulation of ice or frost on structural members comprising flow channels through the ice condenser ≤ [0.38] inches thick.	18 months
SR 3.6.9.6	Visually inspect, for detrimental structural wear, cracks, corrosion or other damage, ≥ 2 ice baskets from each azimuthal group of bays. See SR 3.6.9.4.	40 months

CROSS-REFERENCES - None.

3.6 CONTAINMENT SYSTEMS

3.6.10 Ice Condenser Doors

LCO 3.6.10 The ice condenser inlet doors, intermediate deck doors, and top deck doors shall be OPERABLE and closed.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ice condenser inlet doors inoperable due to being physically restrained from opening.	A.1 Restore inlet doors to OPERABLE status.	12 hours
B. -------NOTE----------- Doors are allowed to be open for Surveillance testing. --------------------- One or more ice condenser door inoperable for reasons other than Condition A or not closed.	B.1 Verify maximum ice bed temperature ≤ [27]°F. AND B.2 Restore ice condenser doors to OPERABLE status and closed positions.	Once per 4 hours 14 days
C. Required Actions and associated Completion Times of Condition B not met.	C.1 Restore ice condenser doors to OPERABLE status and closed positions.	48 hours
D. Required Actions and associated Completion Times of Condition A or C not met.	D.1 Be in Mode 3. AND D.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	Verify all inlet doors indicate closed by the Inlet Door Position Monitoring System.	12 hours
SR 3.6.10.2	Verify, by visual inspection, each intermediate deck door is closed and not impaired by ice, frost, or debris.	7 days
SR 3.6.10.3	Verify, by visual inspection, each inlet door not impaired by ice, frost, or debris.	18 months
SR 3.6.10.4	Demonstrate torque required to cause each inlet door to begin to open is ≤[675] inch-lbs.	18 months
SR 3.6.10.5	Perform a torque test on a sampling of ≥ 25% of the inlet doors.	18 months
SR 3.6.10.6	Verify for each intermediate deck door: a. No visual evidence of structural deterioration, b. Free movement of the intermediate deck vent assemblies, c. Free movement of the door.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.7	Verify, by visual inspection, each top deck door: a. Is closed and in place, and b. Has no condensation, frost, ice formed on the doors or debris that would restrict their opening. c. Has free movement of the top deck vent assemblies.	92 days

CROSS-REFERENCES - None.

3.6 CONTAINMENT SYSTEMS

3.6.11 Divider Barrier Integrity

LCO 3.6.11 Divider barrier integrity shall be maintained.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more personnel access doors or equipment hatches open or inoperable, other than for personnel transit entry.	A.1 Restore personnel access doors and equipment hatches to OPERABLE status and closed positions.	12 hours
B. Divider barrier seal inoperable.	B.1 Restore seal to OPERABLE status.	12 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.11.1	Verify, by visual inspection, all personnel access doors and equipment hatches between upper and lower containment compartments closed.	Prior to entering MODE 4 from MODE 5
SR 3.6.11.2	Verify, by visual inspection, that the seals and sealing surfaces of each personnel access door and equipment hatch have: a. No detrimental misalignments, b. No cracks or defects in the sealing surfaces, and c. No apparent deterioration of the seal material.	Prior to final closure after each opening AND -----Note----- Only required for seals made of resilient materials. ------------ 10 years
SR 3.6.11.3	Verify, by visual inspection, each personnel access door or equipment hatch that has been opened for personnel transit entry is closed.	After each opening

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.4	Remove 2 divider barrier seal test coupons and verify: a. Both test coupons' tensile strength ≥ [60] psid. b. If failure occurs in a., 4 coupons will be tested to 30 psid with no failure. c. If failures occurs in b., 5 coupons will be sent to the manufacturer for LOCA environment simulation testing to 15 psid.	18 months
SR 3.6.11.5	Visually inspect ≥ [95]% of the divider barrier seal length, and verify: a. Seal and seal mounting bolts are properly installed, and b. Seal material shows no evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearance.	18 months

CROSS REFERENCES - None.

3.6 CONTAINMENT SYSTEMS

3.6.12 Hydrogen Analyzer System

LCO 3.6.12 Two hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen analyzer inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable --------------------------- A.1 Restore hydrogen analyzer to OPERABLE status.	30 days
B. Two hydrogen analyzers inoperable.	B.1 Restore one hydrogen analyzer to OPERABLE status.	7 days
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.12.1	Perform CHANNEL CHECK for each hydrogen analyzer.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.2	Perform ANALOG CHANNEL OPERATIONAL TEST for each hydrogen analyzer.	31 days
SR 3.6.12.3	Perform CHANNEL CALIBRATION	92 days

CROSS REFERENCES - None.

3.6 CONTAINMENT SYSTEMS

3.6.13 Hydrogen Recombiner System

LCO 3.6.13 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombiner inoperable.	-------------NOTE----------- LCO 3.0.4 is not applicable ---------------------------- A.1 Restore hydrogen recombiner to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Restore one hydrogen recombiner to OPERABLE status.	7 days
C. Required Actions not met within required Completion Time.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Perform a system functional test for each recombiner.	6 months
SR 3.6.13.2	Perform CHANNEL CALIBRATION of hydrogen recombiner instrumentation and control circuits.	18 months
SR 3.6.13.3	Visually examine each hydrogen recombiner enclosure and ensure there is no evidence of abnormal conditions.	18 months
SR 3.6.13.4	Perform a resistance-to-ground test of each heater phase.	18 months

CROSS REFERENCES - None

3.6 CONTAINMENT SYSTEMS

3.6.14 Hydrogen Mitigation System

LCO 3.6.14 Two Hydrogen Mitigation trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen mitigation train inoperable.	------------NOTE----------- LCO 3.0.4 is not applicable ---------------------------	
	A.1 Restore hydrogen mitigation train to OPERABLE status.	7 days
	OR	
	A.2 Perform SR 3.6.14.1 on OPERABLE train.	Once per 7 days
B. Two hydrogen mitigation trains inoperable.	B.1 Restore one hydrogen mitigation train to OPERABLE status.	7 days
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.14.1	-------------------NOTE------------------- Inoperable igniters shall not be on redundant circuits which provide coverage for the same containment region. -- Energize both Hydrogen Mitigation System power supply breakers and demonstrate that at least 32 of 34 igniters are energized.	 92 days
SR 3.6.14.2	Energize each hydrogen igniter and demonstrate that adequate voltage and amperes exist to produce a temperature $\geq$ [1700]°F.	18 months

CROSS-REFERENCES - None.

3.6 CONTAINMENT SYSTEMS

3.6.15 Containment Recirculation Drains

LCO 3.6.15 The ice condenser floor drains and the refueling canal drains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ice condenser floor drain inoperable.	A.1 Restore ice condenser floor drains to OPERABLE status.	12 hours
B. One refueling canal drain inoperable.	B.1 Restore refueling canal drain to OPERABLE status.	12 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.15.1	Verify by visual inspection that: a. Each refueling canal drain plug is removed, b. Each refueling canal drain is not obstructed by debris, and c. No debris is present in the upper compartment or refueling canal that could obstruct the refueling canal drain.	92 days <u>AND</u> Prior to entering MODE 4 from MODE 5 after each partial or complete fill of the canal
SR 3.6.15.2	Verify for each ice condenser floor drain that: a. The valve gate opening is not impaired by ice, frost, or debris, b. The valve seat shows no evidence of damage, c. The valve gate opening force is $\leq$ [100] lbs, and d. The drain line from the ice condenser floor to the lower compartment is unrestricted.	18 months

CROSS REFERENCES - None

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves

LCO 3.7.1 Main Steam Safety Valves (MSSVs) shall be OPERABLE as specified in Tables 3.7.1-1 and 3.7.1-2.

APPLICABILITY: MODE 1, 2, AND 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required MSSVs OPERABLE.	A.1 Verify at least [2] MSSVs per steam generator are OPERABLE.	4 hours
	AND	
	A.2.1 Restore MSSVs to OPERABLE status.	4 hours
	OR	
	A.2.2.1 Reduce power to ≤ the applicable RATED THERMAL POWER listed in Table 3.7.1-1.	4 hours
	AND	
	A.2.2.2 Reduce the Power Range Neutron Flux-High trip setpoint to ≤ the maximum allowable value listed in Table 3.7.1-1.	8 hours
B. Required Actions and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	-----------------NOTE----------------- SR 3.0.1 and SR 3.0.4 are not applicable for entry into and operation in MODE 3 for the performance of this Surveillance. --------------------------------------- Demonstrate the MSSVs lift setpoints in accordance with the Inservice Inspection and Testing Program.	In accordance with the Inservice Inspection and Testing Program.

CROSS-REFERENCES

TITLE	NUMBER
Inservice Inspection and Testing Program	5.9.14

TABLE 3.7.1-1 (Page 1 of 1)

Power Range Neutron Flux--High Trip
Setpoint Versus OPERABLE Main Steam Safety Valves

Minimum Number of MSSVs per Steam Generator Required OPERABLE	Maximum Applicable RATED THERMAL POWER (%)	Maximum Allowable Power Range Neutron Flux--High Trip Setpoint Ceiling (% of RATED THERMAL POWER) *
5	[>87]	[111.4]
4	[<87, ≥65]	[87]
3	[<65, ≥43]	[65]
2	[<43]	[43]

* NOT applicable in MODE 3.

TABLE 3.7.1-2 (Page 1 of 1)

Main Steam Safety Valve Lift Settings

Valve Number				Lift Setting
SG#1	SG#2	SG#3	SG#4	PSIG, [± 1%]
1-522	1-517	1-512	1-527	[1224]
1-523	1-518	1-513	1-528	[1215]
1-524	1-519	1-514	1-529	[1205]
1-525	1-520	1-515	1-530	[1195]
1-526	1-521	1-516	1-531	[1185]

-------------------------NOTE--------------------------
The lift setting pressure shall correspond to the ambient conditions of the valve at nominal operating temperature and pressure.

3.7 PLANT SYSTEMS

3.7.2 Main Steam Line Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with MSIVs or associated bypass valves open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV or associated bypass valve inoperable.	A.1 Restore MSIV or associated bypass valve to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Close inoperable valve.	6 hours
C. Required actions and associated Completion Times of Condition B not met.	C.1 Be in MODE 3. OR C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------NOTE------------------- SR 3.0.1 and SR 3.0.4 are not applicable for entry into and operation in MODE 3 with MSIVs open, for performance of this surveillance. -- Demonstrate MSIV closure time is ≤ [5] seconds and bypass MSIV ≤ [10] seconds.	In accordance with the Inservice Inspection and Testing Program.

CROSS-REFERENCES

TITLE	NUMBER
Containment Isolation Valves	3.6.6
Inservice Inspection and Testing Program	5.9.14

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Regulation and Isolation Valves (MFRVs and MFIVs)

LCO 3.7.3 [Eight] MFRVs and [eight] MFIVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 with MFRVs and MFIVs open and not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MFRV or MFIV in one or more flow paths inoperable.	A.1 Restore inoperable MFRV or MFIV to OPERABLE status.	72 hours
	OR	
	A.2 Close or isolate inoperable MFRV or MFIV.	72 hours
B. One MFRV and MFIV inoperable in the same flow path.	B.1 Restore at least one MFRV or MFIV to OPERABLE status.	8 hours
	OR	
	B.2 Close inoperable MFRV and MFIV or otherwise isolate each affected flow path.	8 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	-------------------NOTE------------------- SR 3.0.4 is not applicable for entry into MODE 3 for performance of the Surveillance. --- Demonstrate the closure time of each MFRV is ≤ [6.5] seconds and MFIV is ≤ [6.5] seconds.	In accordance with the Inservice Inspection and Testing Program

CROSS-REFERENCES

TITLE	NUMBER
Containment Isolation Valves	3.6.6
Inservice Inspection and Testing Program	5.9.14

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater System

LCO 3.7.5 The Auxiliary Feedwater System (AFW) shall be OPERABLE.

-----------------------NOTE-------------------------
Only one motor-driven pump is required in MODE 4.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 when steam generator is relied on for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine-driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days
B. One AFW subsystem inoperable for reasons other than Condition A.	B.1 Restore AFW subsystem to OPERABLE status.	72 hours
C. Two AFW subsystem inoperable. OR Required Action and associated Completion Times for Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Three AFW subsystems inoperable.	----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until at least one AFW train is restored to OPERABLE status. ------------------------	
	D.1 Initiate action to restore one AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power-operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days
SR 3.7.5.2	----------------NOTE------------------ SRs 3.0.1 and 3.0.4 are not applicable for entry into and operation in MODE 3 for purposes of testing the turbine-driven AFW pumps. ---------------------------------------	
	Demonstrate each AFW pump's developed head at the flow test point is $\geq$ the required developed head.	31 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	Demonstrate each automatic valve actuates to its correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	----------------NOTE------------------- SRs 3.0.1 and 3.0.4 are not applicable for entry into and operation in MODE 3 for purposes of testing the turbine-driven AFW pumps. --------------------------------------- Demonstrate each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Demonstrate the required AFW flow paths from the condensate storage tank (CST) to the steam generator through one of the AFW trains.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for >30 days

CROSS-REFERENCES

TITLE	NUMBER
1. Containment Isolation Valves	3.6.6
2. Condensate Storage Tank	3.7.6
3. Inservice Inspection and Testing Program	5.9.14

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST level shall be OPERABLE with a contained water volume of ≥ [200,000] gallons.

APPLICABILITY: MODES 1, 2, and 3,

MODE 4 when steam generator is relied on for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST level not within limits.	A.1 Restore CST level to within limits. OR A.2.1 Verify OPERABILITY of ERCW System backup water supply. AND A.2.2 Restore CST level to within limits.	4 hours 4 hours AND Once per 12 hours thereafter 7 days
B. Required Actions and associated Completion Times not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 without reliance on steam generator for heat removal.	6 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify CST level $\geq$ [200,000] gallons.	12 hours

CROSS-REFERENCES

TITLE	NUMBER
1. Auxiliary Feedwater System	3.7.5

3.7 PLANT SYSTEMS

3.7.7 Secondary Coolant Specific Activity

LCO 3.7.7 The Specific Activity of the secondary coolant shall be $\leq$ [0.10] μCi/g 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. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Demonstrate the Specific Activity of the secondary coolant is $\leq$ [0.10] μCi/g DOSE EQUIVALENT I-131.	[31] days

CROSS-REFERENCES

TITLE	NUMBER
1. RCS Operational Leakage	3.4.12
2. RCS Specific Activity	3.4.15

3.7 PLANT SYSTEMS

3.7.8 Component Cooling Water System

LCO 3.7.8 Two Component Cooling Water System (CCS) trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCS train inoperable.	A.1 Restore CCS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Two CCS trains inoperable.	C.1 Be in MODE 4. AND C.2 Initiate action to place unit in MODE 5.	12 hours 13 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify each CCS manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days
SR 3.7.8.2	Demonstrate each CCS automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal.	18 months
SR 3.7.8.3	Demonstrate each CCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.8.4	Verify that the alternate feeder breaker to the C-S pump is open.	Once per 7 days when the C-S pump is required OPERABLE

CROSS-REFERENCES

TITLE	NUMBER
1. Containment Isolation Valves	3.6.6
2. RCS Loops - MODE 4	3.4.6
3. Essential Raw Cooling Water System	3.7.9
4. ECCS Trains - Operating (T_{avg} $\geq$350°F)	3.5.2
5. ECCS Trains - Shutdown (T_{avg} <350°F)	3.5.3
6. Inservice Inspection and Testing Program	5.9.14

3.7 PLANT SYSTEMS

3.7.9 Essential Raw Cooling Water (ERCW) System

LCO 3.7.9 Two ERCW system trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ERCW system train inoperable.	A.1 Restore ERCW system train to OPERABLE status.	72 hours
B. Required Action associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Two ERCW system trains inoperable.	C.1 Be in MODE 4. AND C.2 Initiate action to place unit in MODE 5.	12 hours 13 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify each ERCW system manual, power-operated, and automatic valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days
SR 3.7.9.2	Demonstrate each ERCW system automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal.	18 months
SR 3.7.9.3	Demonstrate each ERCW system pump starts automatically on an actual or simulated actuation signal.	18 months

CROSS-REFERENCES

TITLE	NUMBER
1. Containment Isolation Valves	3.6.6
2. Containment Spray System	3.6.3
3. Containment Air Temperature	3.6.8
4. Component Cooling Water System	3.7.8
5. A.C. Sources - Operating	3.8.1
6. Inservice Inspection and Testing Program	5.9.14
7. Control Room Emergency Air Temperature Control (HVAC) System	3.7.14

3.7 PLANT SYSTEMS

3.7.10 Ultimate Heat Sink

LCO 3.7.10 The Ultimate Heat Sink shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ultimate Heat Sink inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify average water temperature of the Ultimate Heat Sink is ≤[85]°F.	[24] hours

CROSS-REFERENCES

TITLE	NUMBER
1. Component Cooling Water System	3.7.8
2. Essential Raw Cooling Water System	3.7.9

3.7 PLANT SYSTEMS

3.7.11 Fuel Storage Pool Water Level

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

APPLICABILITY: When irradiated fuel assemblies are in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool water level not within limit.	-------------NOTE----------- LCO 3.0.3 and 3.0.4 are not applicable. ----------------------------	
	A.1 Suspend movement of fuel assemblies in fuel storage pool.	Immediately
	AND	
	A.2 Initiate action to restore the fuel storage pool water level.	Immediately

SURVEILLANCE REQUIREMENTS

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

CROSS-REFERENCES - None.

3.7 PLANT SYSTEMS

3.7.12 Atmospheric Relief Valves

LCO 3.7.12 Four Atmospheric Relief Valve (ARV) lines shall be OPERABLE.

APPLICABILITY: MODE 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ARV line inoperable.	------------NOTE------------- LCO 3.0.4 is not applicable. ----------------------------- A.1 Restore ARV line to OPERABLE status.	 7 days
B. More than one ARV line inoperable.	B.1 Restore at least three ARV lines to OPERABLE status.	24 hours
C. Required Actions and associated Completion Times for Conditions A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Perform one complete cycle of the ARV.	18 months

CROSS-REFERENCES - None.

3.7 PLANT SYSTEMS

3.7.13 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.13 Two Control Room Emergency Ventilation System (CREVS) trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODES 1, 2, 3, and 4. OR Two CREVS trains inoperable in MODES 1, 2, 3, and 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODES 5 and 6 or during movement of irradiated fuel.	C.1 Place OPERABLE train in the [isolation] mode. OR C.2.1 Suspend CORE ALTERATIONS. AND (continued)	7 days Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.2 Suspend positive reactivity additions. AND C.2.3 Suspend movement of irradiated fuel.	Immediately Immediately
D. Two CREVS trains inoperable in MODES 5 or 6 or during movement of irradiated fuel.	D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend positive reactivity additions. AND D.2.3 Suspend movement of irradiated fuel.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each CREVS train for $\geq$ 15 minutes.	31 days
SR 3.7.13.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program.
SR 3.7.13.3	Demonstrate each CREVS train actuates on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.4	Demonstrate one CREVS train can maintain a positive pressure of ≥ [0.125] inches water gauge relative to [the outside atmosphere] during the [isolation] mode of operation at a flow rate of [4000± 10%] cfm recirculation air and ≤ [1000] cfm of fresh air.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Ventilation Filter Testing Program	5.9.13

3.7 PLANT SYSTEMS

3.7.14 Control Room Emergency Air Temperature Control (HVAC) System

LCO 3.7.14 Two Control Room Emergency Air Temperature Control (HVAC) System trains shall be OPERABLE.

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

----------------NOTE----------------
LCO 3.0.4 is not applicable for entry into MODES 5 or 6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Control Room Emergency Air Temperature Control (HVAC) System train inoperable.	A.1 Restore Control Room Emergency Air Temperature Control (HVAC) system train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODES 1, 2, 3, or 4.	B.1 Be in MODE 3.	6 hours
	AND	
	B.1 Be in MODE 5.	36 hours
C. Required Action and associated Completion Times of Condition A not met in MODES 5 or 6 or during movement of irradiated fuel.	C.1 Place OPERABLE Control Room Emergency Air Temperature Control System train in operation.	7 days
	OR	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend CORE ALTERATIONS. AND C.2.2 Suspend positive reactivity additions. AND C.2.3 Suspend movement of irradiated fuel.	Immediately Immediately Immediately
D. Two Control Room Emergency Air Temperature Control System trains inoperable in MODES 5 and 6 or during movement of irradiated fuel.	D.1 Suspend CORE ALTERATIONS. AND D.2 Suspend positive reactivity additions. AND D.3 Suspend movement of irradiated fuel.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Demonstrate each train of the Control Room Emergency Air Temperature Control (HVAC) System operates for $\geq$ 15 minutes.	31 days

CROSS-REFERENCES

TITLE	NUMBER
Control Room Emergency Ventilation System	3.7.13

3.7 PLANT SYSTEMS

3.7.15 Auxiliary Building Gas Treatment System (ABGTS)

LCO 3.7.15 Two ABGTS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4 and,
During movement of irradiated fuel in the fuel handling area.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ABGTS train inoperable.	A.1 Restore ABGTS train to OPERABLE status.	7 days
B. Required Actions and associated Completion Times of Condition A not met in MODES 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Actions and associated Completion Times for Condition A or B not met during movement of irradiated fuel in the fuel handling area.	C.1 Place OPERABLE ABGTS train in operation. OR C.2 Suspend movement of irradiated fuel in the fuel handling area.	Immediately Immediately
D. Two ABGTS trains inoperable during movement of irradiated fuel in the fuel handling area.	------------NOTE------------ LCO 3.0.3 is not applicable. --------------------------- D.1 Suspend movement of irradiated fuel in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Operate each ABGTS train for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.15.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program.
SR 3.7.15.3	Demonstrate each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.15.4	Demonstrate one ABGTS train can maintain a negative pressure $\geq$ [-0.25] inches water gauge relative to atmospheric pressure during the [post-accident] mode of operation at a flow rate of [9000 ±10%] cfm while maintaining a vacuum relief flow >[2000] cfm.	18 months
SR 3.7.15.5	Demonstrate each ABGTS automatic damper actuates to its correct position on an actual or simulated actuation signal.	18 months

CROSS-REFERENCES

TITLE	NUMBER
Ventialtion Filter Testing Program	5.9.13

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC Power Sources shall be OPERABLE:

a. Two circuits between the offsite transmission network and the onsite Class 1E Power Distribution system, and

b. Four diesel generators.

---------------------NOTE---------------------
The C-S diesel generator may be substituted for any of the required diesel generators.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for remaining required offsite circuit. AND A.2 Restore required offsite circuit to OPERABLE status.	1 hour AND Once per 8 hours thereafter 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required offsite circuit inoperable. AND No offsite power available to one 6.9 kV Shutdown Board. AND Any required feature powered from the 6.9 kV Shutdown Boards that has [have] offsite power available inoperable. OR The turbine-driven auxiliary feedwater pump inoperable.	B.1 Perform SR 3.8.1.1 for remaining required offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	B.2.1 Restore required offsite circuit to OPERABLE status.	24 hours
	OR	
	B.2.2 Restore offsite power availability to all 6.9 kV Shutdown Board.	24 hours
	OR	
	B.2.3.1 Restore required features to OPERABLE status.	24 hours
	AND	
	B.2.3.2 --------NOTE---------- Required Action B.2.3.2 is only required in MODES 1, 2, 3, and MODE 4 when auxiliary feedwater is being used for plant shutdown and startup. ---------------------- Restore turbine-driven auxiliary feedwater pump to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------NOTE---------- Required Actions C.2.1 or C.2.2 shall be completed if this condition is entered. --------------------- One required diesel generator inoperable.	C.1 Perform SR 3.8.1.1 for required offsite circuits.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	C.2.1 Determine absence of common cause for the diesel generator inoperability that would make the other diesel generators inoperable.	[24] hours
	<u>OR</u>	
	C.2.2 Perform SR 3.8.1.2 for OPERABLE diesel generators.	[24] hours
	<u>AND</u>	
	C.3 Restore required diesel generator to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. ---------NOTE---------- Required Actions D.2.1 or D.2.2 shall be completed if this condition is entered. ----------------------- One required diesel generator inoperable. AND Any required feature powered from the OPERABLE diesel generators inoperable. OR The turbine-driven auxiliary feedwater pump inoperable.	D.1	Perform SR 3.8.1.1 for required offsite circuits.	1 hour AND Once per 8 hours thereafter.
	AND		
	D.2.1	Determine absence of common cause for diesel inoperability that would make the other diesel generators not OPERABLE. OR	[16] hours
	D.2.2	Perform SR 3.8.1.2 for OPERABLE diesel generators.	[16] hours
	AND		
	D.3.1	Restore required diesel generators to OPERABLE status. OR	24 hours
	D.3.2.1	Restore required features to OPERABLE status. AND	24 hours
		(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.3.2.2 --------NOTE--------- Required Action D.3.2.2 is only required in MODES 1, 2, 3, and in MODE 4 when auxiliary feedwater is being used for plant shutdown and startup. ---------------------	
	Restore turbine-driven auxiliary feedwater pump to OPERABLE status.	24 hours
E. ---------NOTE-------- Required Action E.2.1 or E.2.2 shall be completed if this condition is entered. --------------------	E.1 Perform SR 3.8.1.1 for remaining required offsite circuit.	1 hour AND Once per 8 hours thereafter
One required offsite circuit inoperable.	AND	
AND One required diesel generator inoperable.	E.2.1 Determine absence of common cause for the diesel inoperability that would make the other diesel generators inoperable.	8 hours
	OR	
	E.2.2 Perform SR 3.8.1.2 for OPERABLE diesel generators.	8 hours
	AND	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	E.3.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> E.3.2 Restore required diesel generator to OPERABLE status.	12 hours 12 hours
F. One required offsite circuit inoperable. <u>AND</u> One required diesel generator inoperable. <u>AND</u> One train de-energized.	F.1 Restore power to the de-energized [safety division] from an offsite circuit or from a diesel generator.	2 hours
G. --------NOTE--------- Required Action G.2.1 or G.2.2 shall be completed if this condition is entered. -------------------- One required offsite circuit inoperable. <u>AND</u> One required diesel generator inoperable. <u>AND</u>	G.1 Perform SR 3.8.1.1 for remaining required offsite circuit. <u>AND</u> G.2.1 Determine absence of common cause for the diesel inoperability that would make the other diesel generators inoperable. <u>OR</u> G.2.2 Perform SR 3.8.1.2 for OPERABLE diesel generators. <u>AND</u> (continued)	1 hour 4 hours 4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. (continued) Any required feature that depends on remaining AC sources inoperable. <u>OR</u> The turbine-driven auxiliary feedwater pump inoperable.	G.3.1 Restore required offsite circuit to OPERABLE status.	8 hours
	<u>OR</u>	
	G.3.2 Restore required diesel generators to OPERABLE status.	8 hours
	<u>OR</u>	
	G.3.3.1 Restore required features to OPERABLE status.	8 hours
	<u>AND</u>	
	G.3.3.2 -------NOTE---------- Required Action G.3.3.2 is only required in MODES 1, 2, 3, and in MODE 4 when auxiliary feedwater is being used for plant shutdown and startup. -------------------- Restore turbine-driven auxiliary feedwater pump to OPERABLE status.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Two required diesel generators inoperable.	H.1 Perform SR 3.8.1.1 for required offsite circuits.	1 hour
	<u>AND</u>	
	H.2 Restore one required diesel generator to OPERABLE status.	2 hours
I. Two required offsite circuits inoperable.	I.1 Restore one required offsite circuit to OPERABLE status.	24 hours
J. Two required offsite circuits inoperable. <u>AND</u> Any required feature that depends on remaining AC sources inoperable. <u>OR</u> The turbine-driven auxiliary feedwater pump inoperable.	J.1 Restore one required offsite circuit to OPERABLE status.	12 hours
	<u>OR</u>	
	J.2.1 Restore required features to OPERABLE status.	12 hours
	<u>AND</u>	
	J.2.2 --------NOTE----------- Required Action J.2.2 is only required in MODES 1, 2, 3, and in MODE 4 when auxiliary feedwater is being used for plant shutdown and startup. ----------------------- Restore turbine-driven auxiliary feedwater pump to OPERABLE status.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. Three or more required AC power sources inoperable.	K.1 Enter LCO 3.0.3	Immediately
L. Required Actions and associated Completion Times not met.	L.1 Be in MODE 3. <u>AND</u> L.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2	-----------------NOTE------------------ 1. This SR does not have to be performed for the C-S diesel generator unless it is being used to satisfy LCO 3.8.1. 2. Performance of SR 3.8.1.5 satisfies this surveillance. 3. Following diesel generator start, warmup procedures such as idling and gradual acceleration may be used as allowed by the manufacturer. [When they are not used, the time, voltage, and frequency tolerances specified in SR 3.8.1.5 must be met.] --	
	Demonstrate each diesel generator starts from standby conditions and achieves the following steady state voltage and frequency: a. Voltage ≥[6210] volts and ≤[7590] volts, and b. Frequency ≥58.8] Hz and ≤[61.2] Hz.	As specified by Table 3.8.1-1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.3	------------------NOTES------------------ 1. Diesel generator 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 diesel generator at a time. --- Demonstrate each diesel generator is synchronized, loaded and operates for ≥ 60 minutes at a load ≥[4400] kW and ≤[4750] kW.	As specified by Table 3.8.1-1
SR 3.8.1.4	Verify pressure in required air start receivers ≥[ ] psig.	31 days
SR 3.8.1.5	-----------------NOTE------------------- 1. Following this Surveillance Requirement, perform SR 3.8.1.3. --- Demonstrate each diesel generator starts from standby condition and achieves the following voltage and frequency in ≤[10] seconds: a. Voltage ≥[6210] volts and ≤[7590] volts, and b. Frequency ≥ [58.8] Hz and ≤[61.2] Hz.	184 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	Demonstrate [automatic/manual] transfer from normal onsite power source of 6.9 kV shutdown board to offsite power source.	[18 months] and <u>not</u> when in MODES 1, 2, 3 and 4
SR 3.8.1.7	Demonstrate each diesel generator rejects a load ≥[600] kW and: a. Following load rejection, the frequency is ≤[61.2] Hz, and b. Within [10] seconds following load rejection, the frequency is ≥[58.8] Hz and ≤[61.2] Hz, and c. Within [10] seconds following load rejection, the voltage is ≥[6210] volts and ≤[7590] volts.	18 months
SR 3.8.1.8	Demonstrate each diesel generator does not trip and voltage is maintained ≤[7866] volts during and following a load rejection ≥[4400] kW and ≤[4750] kW.	[18 months] and <u>not</u> when in MODES 1, 2, 3 and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	Demonstrate on an actual or simulated loss of offsite power signal: a. De-energization of emergency, buses, b. Load shedding from emergency buses, and c. Diesel generator auto-starts from standby condition and: 1. Energizes permanently-connected loads in ≤[10] seconds, 2. Energizes auto-connected shutdown loads through load sequencer, 3. Maintains steady state voltage ≥[6219] volts and ≤[7590] volts. 4. Maintains steady state frequency ≥[58.8] Hz and ≤[61.2] Hz, and 5. Supplies permanently-connected and auto-connected shutdown loads for ≥[5] minutes.	[18 months] and <u>not</u> when in MODES 1, 2, 3, and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	Demonstrate on an actual or simulated loss of offsite power signal with a delayed [Engineered Safety Features (ESF)] actuation signal that the blackout (BO) load sequencing resets to the [ESF-BO] load sequencing. After the last load step, simulate loss of offsite power and demonstrate: a. De-energization of emergency buses, b. Load shedding from emergency buses, c. Diesel generator from ready to load condition:	[36 months] on a STAGGERED TEST BASIS with SR 3.8.1.11 and <u>NOT</u> when in MODES 1, 2, 3 and 4
SR 3.8.1.11	Demonstrate on an actual or simulated [Engineered Safety Features (ESF)] signal, each diesel generator auto-starts from standby condition and: a. Achieves voltage ≥[6210] volts and ≤[7590] volts in ≤[10] seconds after auto-start and during test, b. Achieves frequency ≥[58.8] Hz and ≤[61.2] Hz in ≤[10] seconds after auto-start and during test, c. Operates for ≥[5] minutes, d. Permanently-connected loads remain energized from the offsite power system, and	[36 months] on a STAGGERED TEST BASIS with SR 3.8.1.10 and <u>not</u> when in MODES 1, 2, 3 and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	Demonstrate on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated [ESF] actuation signal: a. De-energization of emergency, buses, b. Load shedding from emergency buses, and c. Diesel generator auto-starts from standby condition and: 1. Energizes permanently-connected loads in ≤[10] seconds, 2. Energizes auto-connected emergency loads through load sequencer, 3. Achieves steady state voltage ≥[6210] volts and ≤[7590] volts. 4. Achieves steady state frequency ≥[58.8] Hz and ≤[61.2] Hz, and 5. Supplies permanently-connected and auto-connected emergency loads for ≥[5] minutes.	[18 months] and <u>not</u> when in MODES 1, 2, 3 and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	Demonstrate each diesel generator's automatic trips are bypassed on [actual or simulated loss of voltage signal on the emergency bus concurrent with and actual or simulated (ESF) actuation signal] except: a. Engine Overspeed b. Generator Differential Current	[18 months] [and <u>not</u> when in MODES 1, 2, 3, and 4]
SR 3.8.1.14	----------------NOTE------------------- Momentary transients outside the load range do not invalidate this test. --------------------------------------- Demonstrate each diesel generator operates for ≥ 24 hours: a. Loaded ≥[4840] kW and ≤[5000] kW for first 2 hours, and b. Loaded ≥[4400]kW and ≤[4750] kW during remaining 22 hours of test.	[18 months] and <u>not</u> when in MODES 1, 2, 3, and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	----------------NOTE---------------- 1. This surveillance shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated ≥ 2 hours at ≥[4400] kW and ≤[4750] kW. 2. Momentary transients outside of load range do not invalidate this test. ------------------------------------ Demonstrate each diesel generator starts and achieves the following voltage and frequency in ≤[10] seconds: a. Voltage ≥[6210] volts and ≤[7590] volts. b. Frequency ≥[58.8] Hz and ≤[61.2] Hz.	18 months
SR 3.8.1.16	Demonstrate each diesel generator: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power. b. Transfers loads to offsite power sources, and c. Returns to ready to load operation.	18 months and not when in MODES 1, 2, 3, and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.17	Demonstrate with a diesel generator operating in test mode, an actual or simulated [ESF] actuation signal overrides the test mode by: a. Returning diesel generator to ready to load operation.	[18 months] [and <u>not</u> when in MODES 1, 2, 3 and 4]
SR 3.8.1.18	Demonstrate the interval between each load block is within $\pm$ [10% of design interval] for each [ESF] load sequencer.	[18 months] [and <u>not</u> when in MODES 1, 2, 3 and 4]
SR 3.8.1.19	Demonstrate all diesel generators achieve the following voltage and frequency in $\leq$[10] seconds when started simultaneously from standby condition: a. Voltage $\geq$[6210] volts and $\leq$[7590] volts, and b. Frequency $\geq$[58.8] Hz and $\leq$[61.2] Hz.	10 years

CROSS-REFERENCS - None.

Table 3.8.1-1 (Page 1 of 1)

Diesel Generator Test Schedule

NUMBER OF FAILURES IN LAST 25 VALID TESTS [a]	TEST FREQUENCY
≤ 3	31 days
≥ 4	7 days[b] (but no less than 24 hours)

(a) Criteria for determining number of failures and valid demands shall be in accordance with Regulatory Position C.2.1 of Regulatory Guide 1.9, Rev. 3, where the number of demands and failures is determined on a per diesel generator basis.

Failure to satisfactorily complete:
SR 3.8.1.4, SR 3.8.1.18, or
SR 3.8.3.1 through SR 3.8.3.10
does not constitute a valid failure and does not require an increased frequency of testing.

(b) This test frequency shall be maintained until seven consecutive failure free start and load-run demands have been performed. If subsequent to the seven failure free tests one or more additional failures occur such that there are again four or more failures in the last 25 tests, the testing interval shall again be reduced as noted above and maintained until seven consecutive failure-free tests have been performed.

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC Power Sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E Power distribution system, and

b. Two diesel generators (either 1A-A and 2A-A or 1B-B and 2B-B).

------------------------NOTE------------------------
The C-S diesel generator may be substituted for any of the required diesel generators.

APPLICABILITY:
a. MODES 5 and 6,
b. When handling irradiated fuel,
c. During movement of heavy loads over irradiated fuel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC Power Sources inoperable.	A.1 Suspend CORE ALTERATIONS.	15 minutes
	AND	
	A.2 Suspend handling of irradiated fuel.	15 minutes
	AND	
	A.3 Suspend movement of heavy loads over irradiated fuel.	15 minutes
	AND	
	A.4 Suspend operations with a potential for draining the reactor vessel.	15 minutes
	AND	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 Suspend operations involving positive reactivity additions. AND A.6.1 Initiate action to restore required AC Power Sources. AND A.6.2 Continue action to restore required AC Power Sources.	15 minutes 15 minutes Until required AC Power Sources are restored

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	Perform: SR 3.8.1.1 through SR 3.8.1.6, SR 3.8.1.8 through SR 3.8.1.11, SR 3.8.1.15, SR 3.8.1.16, and SR 3.8.1.18.	As specified by applicable SRs

CROSS-REFERENCES - None.

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel and Lubricating Oil

LCO 3.8.3 The diesel fuel oil subsystem shall be OPERABLE and lubricating oil inventory shall be sufficient for each required diesel generator.

APPLICABILITY: When associated diesel generator is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel level low in one or more skid-mounted day tanks.	A.1 Restore fuel level in skid-mounted day tanks.	[1] hour
B. Fuel transfer capability inoperable for one or more diesel generators	B.1 Restore fuel transfer capability to OPERABLE status.	[4] hours
C. Fuel level low in one or more 7-day tanks.	C.1 Restore fuel level in 7-day tank(s).	[24] hours
D. Lubricating oil inventory insufficient.	D.1 Restore lubricating oil inventory.	[24] hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Fuel properties, except for accelerated stability testing, do not meet limits in one or more 7-day storage tanks.	E.1 Restore fuel properties to within limits.	[72] hours
F. Fuel accelerated stability testing properties do not meet limits in one or more 7-day storage tanks.	F.1 Restore fuel properties to within limits.	[30] days
G. Required Actions and associated Completion Times not met. <u>OR</u> Diesel fuel subsystem inoperable for reasons other than Condition A, B, C, E, or F.	G.1 Declare associated diesel generator(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each skid-mounted day fuel tank contains ≥[250] gallons of fuel.	24 hours
SR 3.8.3.2	Verify each 7-day fuel storage tank contains ≥[60,000] gallons of fuel.	31 days
SR 3.8.3.3	Verify lubricating oil inventory is ≥[331] gallons of oil per engine.	31 days
SR 3.8.3.4	Demonstrate that the flash point, gravity, viscosity, and appearance of new fuel are within limits when tested in accordance with applicable ASTM standards.	Immediately prior to addition of new fuel to storage tank(s).
SR 3.8.3.5	Demonstrate that the properties of new fuel, other than those in SR 3.8.3.4, are within applicable ASTM limits.	Within 31 days following addition of new fuel to the storage tanks.
SR 3.8.3.6	Demonstrate that the total particulate in stored fuel is < 10 mg/liter when tested in accordance with applicable ASTM standards.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.7	Check for and remove accumulated water from each 7-day storage tank.	31 days
SR 3.8.3.8	Check for and remove accumulated water from each skid-mounted day tank.	31 days
SR 3.8.3.9	Demonstrate the fuel transfer system operates to [automatically] transfer fuel from 7-day storage tank(s) to the engine mounted tank.	[92] days
SR 3.8.3.10	For the fuel subsystem: a. Drain each 7-day fuel storage tank. b. Remove the sediment from the 7-day storage tank. c. Clean the 7-day storage tank.	10 years

CROSS-REFERENCES

TITLE	NUMBER
AC Sources - Operating	3.8.1
AC Sources - Shutdown	3.8.2

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Two trains of DC Power Sources shall be OPERABLE in accordance with Table 3.8.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC Power Source inoperable.	A.1 Restore DC Power Source to OPERABLE status.	2 hours
B. Required Actions and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.4.1	Verify each battery terminal voltage ≥ [126] volts on float charge.	7 days
SR 3.8.4.2	Verify no visible corrosion at terminals and connectors. OR Demonstrate connection resistance of terminals and connectors is ≤[150 E-6] ohms.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.3	Verify the cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
SR 3.8.4.4	Verify the cell-to-cell terminal connections are clean, tight, free of visible corrosion, and coated with anti-corrosion material.	18 months
SR 3.8.4.5	Demonstrate the resistance of each cell-to-cell and terminal connection is ≤[150 E-6] ohms.	18 months
SR 3.8.4.6	Demonstrate each battery charger will supply ≥[200] amperes at ≥[125] volts for ≥[4] hours.	18 months
SR 3.8.4.7	--------------NOTE------------------ One per 60-month interval, SR 3.8.4.8 may be performed in lieu of this surveillance. ------------------------------------- Demonstrate battery capacity is adequate to supply and maintain in an OPERABLE status, the required emergency loads for the design duty cycle, when subjected to a battery service test.	18 months and <u>not</u> when in MODES 1, 2, 3, and 4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.8	Demonstrate the battery capacity is ≥ [ ]% of the manufacturer's rating when subjected to a performance discharge test.	60 months and not when in MODES 1, 2, 3, and 4 AND ----NOTE---- Only applicable when battery shows degradation or has reached ≥[ ]% of the expected life ------------- 18 months and not when in MODES 1, 2, 3 and 4

CROSS-REFERENCES

TITLE	NUMBER
Distribution Systems - Operating	3.8.7
Battery Cell Parameters	3.8.6

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 One train of DC Power Sources shall be OPERABLE in accordance with Table 3.8.7-1.

APPLICABILITY:
a. MODES 5 and 6,
b. When handling of irradiated fuel,
c. During movement of heavy loads over irradiated fuel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required DC Power Source inoperable.	A.1 Suspend CORE ALTERATIONS.	15 minutes
	AND	
	A.2 Suspend handling of irradiated fuel.	15 minutes
	AND	
	A.3 Suspend movement of heavy loads over irradiated fuel.	15 minutes
	AND	
	A.4 Suspend operations with a potential for draining the reactor vessel.	15 minutes
	AND	
	A.5 Suspend operations involving positive reactivity additions.	15 minutes
	AND	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6.1 Initiate action to restore one DC Power Source to OPERABLE status.	15 minutes
	AND	
	A.6.2 Continue action to restore one DC Power Source to OPERABLE status.	Until one DC Power Source restored

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	Perform SR 3.8.4.1 through SR 3.8.4.8.	As specified by SR 3.8.4.1 through SR 3.8.4.8

CROSS-REFERENCES

TITLE	NUMBER
Distribution Systems - Shutdown	3.8.8
AC Sources-Shutdown	3.8.2
Battery Cell Parameters	3.8.6

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 DC Power Source cell parameters shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC Power Sources are required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more cells in one or more batteries not within limits.	A.1 Demonstrate the pilot cells' electrolyte level and float voltage meet Category C allowable values.	24 hours
	AND	
	A.2 Demonstrate the parameters in Table 3.8.6-1 meet Category C allowable values.	24 hours
	AND	
	A.3 Restore the parameters to Category A and B limits of Table 3.8.6-1.	31 days
B. Required Actions and associated Completion Times not met.	B.1 Declare associated DC Power Source inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify the parameters in Table 3.8.6-1 meet the Category A limits.	7 days
SR 3.8.6.2	Verify the parameters in Table 3.8.6-1 meet the Category B limits.	92 days AND Once within 72 hours after a battery discharge below [105] volts AND Once within 72 hours after a battery overcharge above [150] volts when system is operational
SR 3.8.6.3	Verify the average electrolyte temperature of the representative cells is $\geq$[60] °F.	92 days

CROSS-REFERENCES

TITLE	NUMBER
DC Sources - Operating	3.8.4
DC Sources - Shutdown	3.8.5

Table 3.8.6-1 (Page 1 of 1)

Battery Cell Parameter Requirements

	CATEGORY A	CATEGORY B	CATEGORY C
Parameter	Limit for each designated pilot cell	Limit for each connected cell	Allowable Value for each connected cell
Electrolyte Level	> Minimum level indication mark, and ≤ 1/4 " above maximum level indication mark	> Minimum level indication mark, and ≤ 1/4 " above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	≥ [2.13] volts (a)	≥ [2.13] volts (a)	>[2.07] volts (a)
Specific Gravity (b)	≥ [1.200] (c)	≥ [1.195] AND Average of all connected cells ≥ [1.205].	Not more than [0.020] below the average of all connected cells AND Average of all connected cells ≥ [1.195]. (c)
Electrolyte Temperature	Average Temperature ≥[60] °F	Average Temperature ≥[60] °F	Average Temperature ≥[60] °F

-----------------------------------NOTES-----------------------------------

(a) May be corrected for average electrolyte temperature.
(b) Corrected for electrolyte temperature and level.
(c) Or battery charging current is <[2] amperes when on float charge.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 The Power Distribution System trains, identified in Table 3.8.7-1, shall be OPERABLE.

---------------------------NOTE---------------------------
Two inverters may be disconnected from their associated DC bus for $\leq$ 24 hours to perform an equalizing charge on associated battery banks, providing:

a. Associated AC vital buses energized, and

b. AC vital buses for other battery banks energized from associated inverters connected to their DC buses.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required buses except AC vital buses in one train not OPERABLE as required in Table 3.8.7-1.	A.1 Restore required DC buses. AND A.2 Restore required AC buses to OPERABLE status.	2 hours 8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One AC vital bus inoperable as required in Table 3.8.7-1.	B.1 Power AC vital bus from its alternate Class 1E voltage source. AND B.2 Restore AC vital bus to OPERABLE staus.	2 hours 24 hours
C. Required Actions and associated Completion Times not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct breaker alignments and voltages for buses required AC and DC Power Distribution Systems identified in Table 3.8.7-1.	7 days

CROSS-REFERENCES - None.

Table 3.8.7-1 (Page 1 of 1)

Power Distribution System Trains

TYPE	VOLTAGE	TRAIN A	TRAIN B
AC emergency buses	6900 VAC	1A-A 2A-A	1B-B 2B-B
	480 VAC	1A1-A 1A2-A 2A1-A 2A2-A	1B1-B 1B2-B 2B1-B 2B2-B
DC buses	125 VDC	Board 1 from Vital Battery Bank 1 Board 3 from Vital Battery Bank 3	Board 2 from Vital Battery Bank 2 Board 4 from Vital Battery Bank 4
AC vital buses	120 VAC	Vital channel 1-I Vital channel 2-I from inverter and DC Board I Vital channel 1-III Vital channel 2-III from inverter and DC Board III	Vital channel 1-II Vital channel 2-II from inverter and DC Board II Vital channel 1-IV Vital channel 2-IV from inverter and DC Board IV

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems - Shutdown

LCO 3.8.8 One Power Distribution System train, identified in Table 3.8.7-1, shall be OPERABLE.

APPLICABILITY:
a. MODES 5 and 6,
b. When handling of irradiated fuel,
c. During movement of heavy loads over irradiated fuel.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No power distribution system train energized.	A.1 Suspend CORE ALTERATIONS.	15 minutes
	AND	
	A.2 Suspend movement of irradiated fuel.	15 minutes
	AND	
	A.3 Suspend movement of heavy loads over irradiated fuel.	15 minutes
	AND	
	A.4 Suspend operations with a potential for draining the reactor vessel.	15 minutes
	AND	
	A.5 Suspend operations involving positive reactivity additions.	15 minutes
	AND	
	(continued)	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6.1 Initiate action to restore required buses to OPERABLE. AND A.6.2 Continue action to restore required buses to OPERABLE status.	15 minutes Until required buses have been restored to OPERABLE status

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Perform SR 3.8.7.1.	7 days

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 The boron concentration of all filled portions of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit provided in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS and positive reactivity additions.	15 minutes
	AND	
	A.2 Initiate boration to restore concentration.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration within limit.	72 hours

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used in the allowable combination of valves to isolate unborated water sources shall be secured in the closed position.

---------------------------NOTE---------------------------
Valves may be opened during planned boron concentration changes or makeup activities when CORE ALTERATIONS are not taking place.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
---------NOTE---------- Required Action A.3 must be completed whenever Condition A is entered. ----------------------		
A. One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Initiate actions to secure valve in closed position.	Immediately
	AND	
	A.3 Perform SR 3.9.1.1. (boron concentration verification).	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.2.1	Verify the combination of valves which isolates unborated water sources is secured in the closed position.	31 days AND Once within 1 hour after a planned dilution or makeup activity

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE with continuous visual indication in the control room and one shall provide audible indication in the containment and in the control room.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable. OR No audible indication in the containment or the control room.	A.1 Suspend CORE ALTERATIONS and positive reactivity additions.	15 minutes
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate actions to restore one channel to OPERABLE staus. AND B.2 Perform surveillance SR 3.9.1.1. (boron concentration verification).	Immediately Within 4 hours AND Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform a CHANNEL CHECK of required source range neutron flux monitors.	12 hours
SR 3.9.3.2	Verify audible indication is provided in the containment and control room.	12 hours
SR 3.9.3.3	Perform ANALOG CHANNEL OPERATIONAL TEST for required source range neutron flux monitors.	7 days

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.4 Containment Building Penetrations

LCO 3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch closed,

b. A minimum of one door in each airlock closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. Closed by an isolation valve, blind flange, manual valve, or equivalent, or
2. Capable of being closed by two OPERABLE Containment Vent Isolation (CVI) System trains:

APPLICABILITY: MODE 6 during CORE ALTERATIONS or movement of irradiated fuel within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CVI System train inoperable.	A.1 Restore inoperable CVI train to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment building penetrations not in required status. OR Two CVI System trains inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1 Place affected containment penetrations in required status. OR B.2 Suspend CORE ALTERATIONS and movement of irradiated fuel within containment.	15 minutes 15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Perform CHANNEL CHECK of each purge exhaust and Containment atmosphere radiation channel.	12 hours
SR 3.9.4.2	Verify each containment building penetration is in its required status.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.9.4.3	Demonstrate Containment Vent Isolation occurs on: a. Manual initiation, and b. A high radiation test signal from each purge exhaust and containment atmosphere radiation monitoring channel.	14 days
SR 3.9.4.4	Perform ANALOG CHANNEL OPERATIONAL TEST for each purge exhaust and containment atmosphere radiation monitoring channel.	31 days
SR 3.9.4.5	Perform CHANNEL CALIBRATION of each purge exhaust and containment atmosphere radiation monitoring channel.	18 months

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal and Coolant Circulation - High Water Level

LCO 3.9.5 At least one Residual Heat Removal (RHR) loop shall be OPERABLE and in operation.

---------------------------NOTE---------------------------
The required RHR loop may be removed from operation or OPERABLE status for ≤ 1 hour per 2-hour period.

APPLICABILITY: MODE 6 with the water level ≥ 23 feet above top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No RHR loop OPERABLE or in operation.	A.1 Suspend all operations involving a reduction in reactor coolant boron concentration.	15 minutes
	AND	
	A.2 Suspend all operations involving an increase in reactor decay heat load.	15 minutes
	AND	
	A.3 Initiate action to restore one RHR loop to OPERABLE status and operation.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify at least one RHR loop operating and circulating reactor coolant.	12 hours

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal and Coolant Circulation - Low Water Level

LCO 3.9.6 Two Residual Heat Removal (RHR) loops shall be OPERABLE and at least one RHR loop shall be in operation.

-----------------------NOTE-----------------------
Prior to initial criticality, only one RHR loop needs to be OPERABLE and in operation and may be removed from operation or OPERABLE status for $\leq$ 1 hour per 2-hour period.

APPLICABILITY: MODE 6 with the water level < 23 feet above top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	15 minutes
	AND	
	A.2 Establish alternate decay heat removal capabilities.	7 days
B. No RHR loop OPERABLE or in operation.	B.1 Suspend all operations involving a reduction in reactor coolant boron concentration.	15 minutes
	AND	
	B.2 Initiate action to restore one RHR loop to OPERABLE status with one loop in operation.	15 minutes
	AND	
	B.3 Initiate action to implement alternate decay heat removal.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify at least one RHR loop operating and circulating reactor coolant.	12 hours
SR 3.9.6.2	Verify, by correct breaker alignment and indicated power availability, that the required RHR loop, which is not in operation, is OPERABLE.	7 days

CROSS-REFERENCES

TITLE	NUMBER
Residual Heat Removal and Coolant Circulation - High Water Level	3.9.5

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ [23] feet above top of reactor vessel flange.

APPLICABILITY: Mode 6 during movement of fuel assemblies and Rod Cluster Control Assemblies (RCCAs) within containment with irradiated fuel in containment.

ACTIONS

CONDITIONS	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of fuel assemblies and RCCAs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify refueling cavity water level within limit.	24 hours

CROSS-REFERENCES - None.

3.9 REFUELING OPERATIONS

3.9.8 Decay Time

LCO 3.9.9 The reactor shall be subcritical for $\geq$ [100] hours.

APPLICABILITY: MODE 6 during CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor subcritical for less than minimum time.	A.1 Suspend CORE ALTERATIONS.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.8.1	Verify reactor subcritical for $\geq$ [100] hours.	Once each shutdown prior to CORE ALTERATIONS

CROSS-REFERENCES - None.

4.0 DESIGN FEATURES

4.1 SITE

4.1.1 Site and Exclusion Boundaries

The site and exclusion boundary area shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone

The low population zone shall be as shown in Figure 4.1-2.

4.1.3 Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents

Information regarding radioactive gaseous and liquid effluents, which allows identification of structures and release points as well as definition of Unrestricted Areas within the Site Boundary that are accessible to Members of the Public, shall be shown in Figure 4.1-1.

4.2 CONTAINMENT

4.2.1 Configuration

The reactor containment building consists of the Shield Building and the Steel Containment Vessel.

4.2.1.1 Shield Building

The Shield Building is a reinforced concrete cylinder supported by a circular base slab and covered at the top with a spherical dome, and it has the following design features:

a. Nominal inside diameter = [125] feet

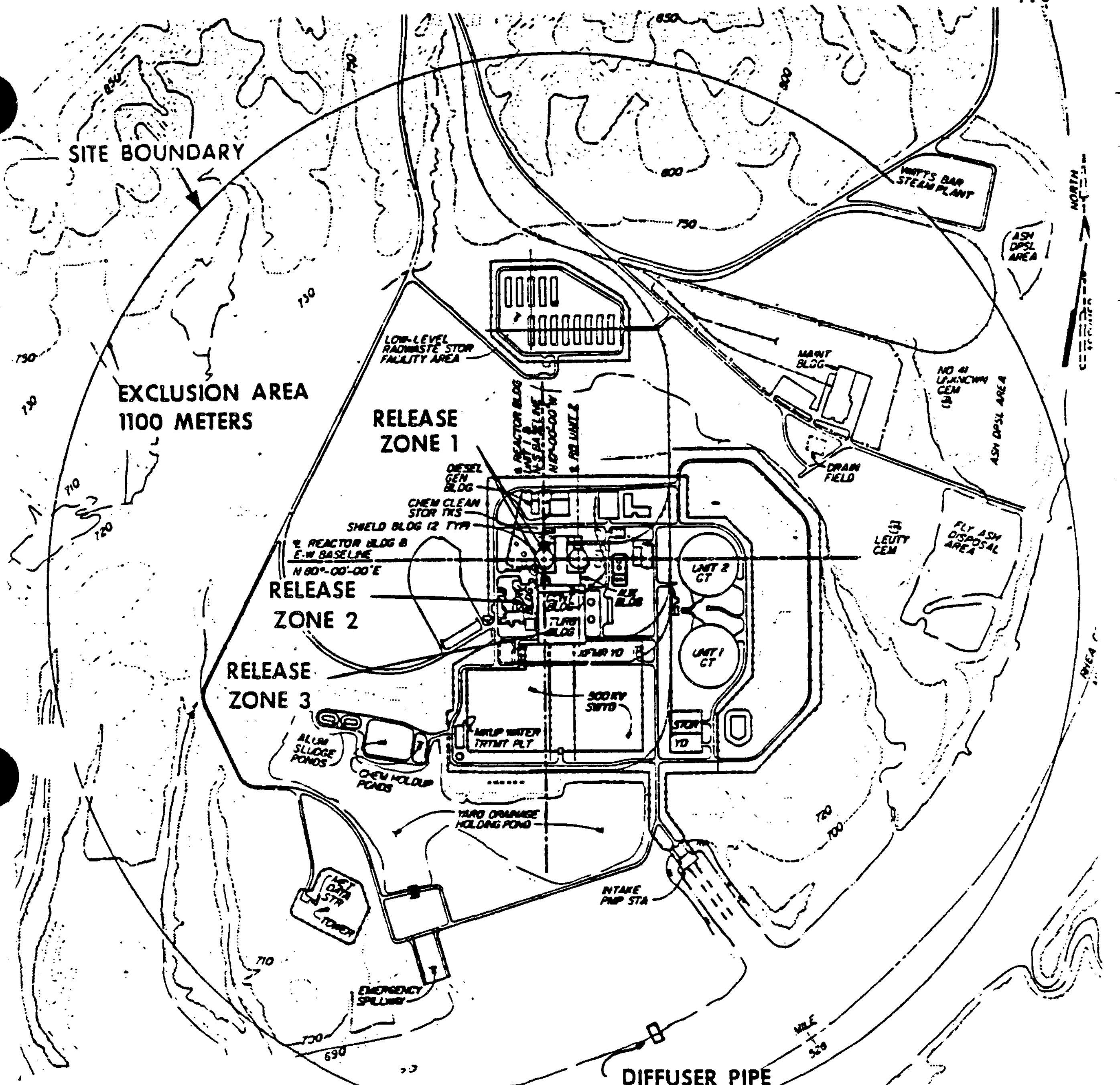
b. Nominal inside height = [174.25] feet

c. Minimum thickness of concrete walls = [3] feet

d. Minimum thickness of concrete roof = [2] feet

e. Minimum thickness of concrete floor pad = [10] feet

(continued)



ROUTINE GASEOUS RELEASES
(elevations above plant grade)

RELEASE ZONE 1
Shield Building Vent. el. 132 ft.
Auxilliary Building Vent. el. 121 ft.

RELEASE ZONE 2
Service Building Vent. el. 26 ft.

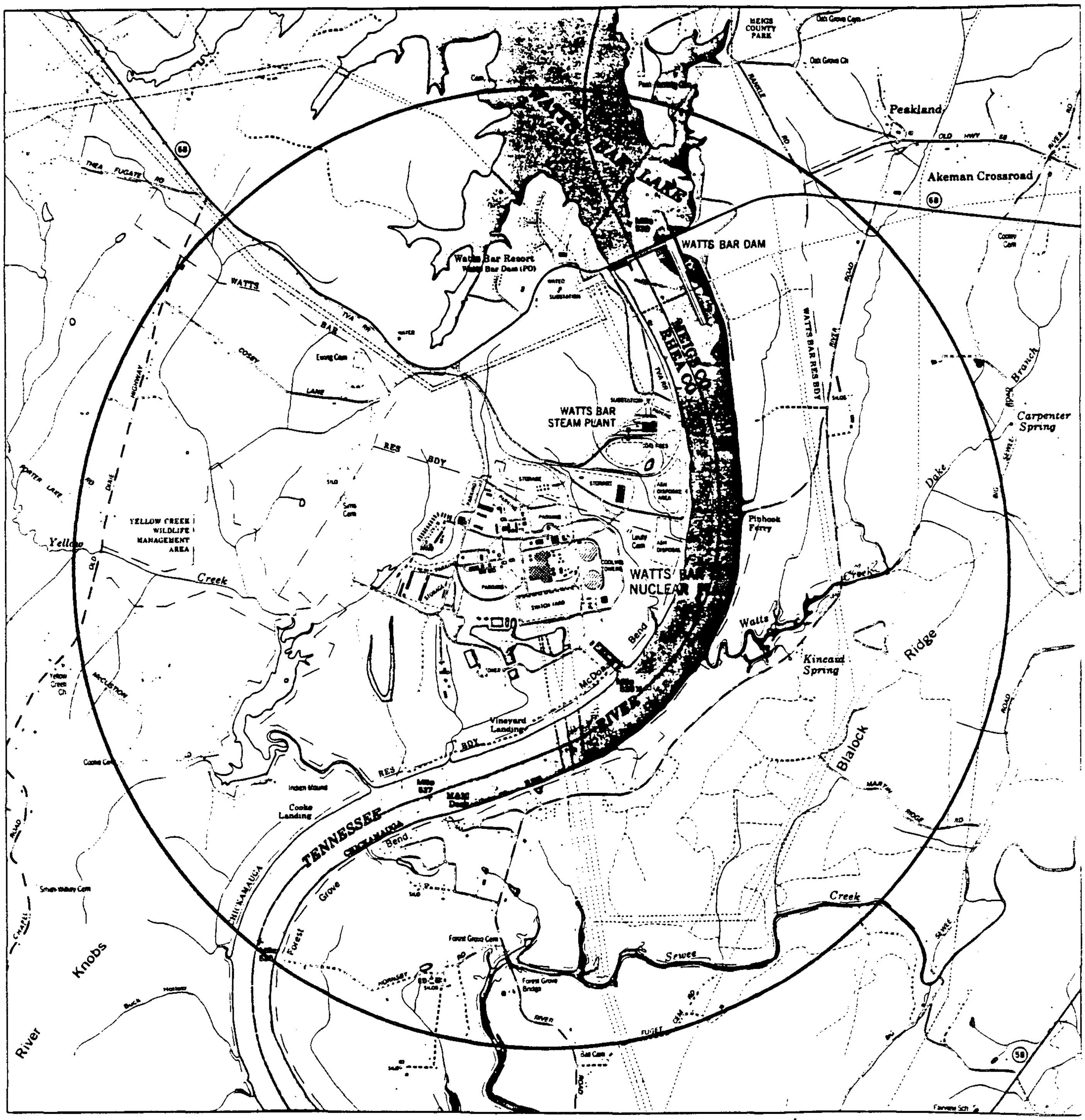
RELEASE ZONE 3
Condenser Air Elector Exhaust Vent. el. 102 ft

ROUTINE LIQUID RELEASE POINT
Diffuser Pipes

NOTE: For the purpose of implementing the Radiological Effluent Technical Specifications, all areas outside the exclusion area radius are considered to be 'Unrestricted Areas'.

0 1100 METERS

Site Boundary/Exclusion Area
Figure 4.1-1



Low Population Zone
Figure 4.1-2

4.0 DESIGN FEATURES (continued)

4.2.1.2 Steel Containment Vessel

The Steel Containment Vessel is a low-leakage, free-standing steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete, and it has the following design features:

a. Nominal inside diameter = [115] feet

b. Nominal height = [170] feet

c. Thickness of the cylindrical portion of the vessel:

   Varies from 1 3/8 inch at the bottom
   to 1-1/2 inch at the spring line

d. Thickness of the hemispherical portion (dome) of the vessel:

   Varies from 13/16 inch to 1 3/8 inch

e. Net free volume = [1,191,414] cubic feet

4.2.2 Design Pressure and Temperature

The reactor containment building is designed and shall be maintained for a maximum internal pressure of [15.0] psig and temperature of [327]°F.

4.2.3 Containment Leak Test Parameters

The containment leak rate test parameters are as follows:

a. Calculated peak DBA interior pressure, P_a = [15.0] psig.

b. Maximum allowable leakage rate, L_a = [0.25]% by weight of the containment air per 24 hours at P_a.

c. A combined leakage rate of $\leq$ [0.6] L_a for all penetrations and valves subject to Type B and C tests, when pressurized to $P_a \geq$ [15.0] psig.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 REACTOR CORE

4.3.1 Fuel Assemblies

The core shall contain [193] fuel assemblies with each fuel assembly nominally containing 264 fuel rods clad with Zircaloy-4, except that substitution of Zircaloy-4 or stainless steel filler rods or open water channels for fuel rods may be made in fuel assemblies if justified by cycle-specific reload analyses using an NRC-approved methodology. Should more than 30 rods in the core, or 10 rods in any assembly, be replaced per refueling, a Special Report describing the number of rods replaced shall be submitted to the Commission pursuant to Specification 5.10.7 within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of [5.0] weight percent U-235.

4.3.2 Control Rod Assemblies

The reactor core shall contain [53] full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal [142] inches of absorber materials consisting of 40 inches Ag-In-Cd and 102 inches of B_4C. The nominal values of Ag-In-Cd absorber material shall be 80% silver, 15% indium and 5% cadmium. All control rods shall be clad with stainless steel tubing.

4.4 REACTOR COOLANT SYSTEM

4.4.1 Design Pressure and Temperature

The reactor coolant system is designed and shall be maintained:

a. In accordance with code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For pressure of [2485] psig, and

c. For a temperature of [650]°F, except for the pressurizer which is [680]°F.

(continued)

4.0 DESIGN FEATURES (continued)

4.4.2 Volume

The total water and steam volume of the reactor coolant system is estimated to be [12,145] cubic feet at a nominal T_{avg} of [588]°F.

4.5 METEOROLOGICAL TOWER LOCATION

4.5.1 The meteorological tower shall be located as shown in Figure 4.1-1.

4.6 FUEL STORAGE

4.6.1 Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to ≤ [0.95] when flooded with unborated water, which includes a conservative allowance of [ ]% Δk/k for uncertainties.

b. A nominal [10.75]-inch center-to-center distance between fuel assemblies placed in the storage racks.

The new fuel pit storage racks are designed and shall be maintained with a nominal 21-inch center-to-center distance between new fuel assemblies such that, on a best estimate basis, k_{eff} will not exceed [0.98], with fuel of the highest anticipated enrichment in place, when aqueous foam moderation is assumed.

4.6.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation [745'- 1-1/2"] feet, Mean Sea Level, USGS datum.

4.6.3 Capacity

The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than [1312] fuel assemblies.

4.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

4.7.1 The components identified in Table 4.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 4.7-1.

(continued)

4.0 DESIGN FEATURES (continued)

TABLE 4.7-1 Page 1 of 1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at 100°F/hr and	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F
	200 cooldown cycles at 100°F/hr	Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F
	200 pressurizer cooldown cycles at 200°F/hr.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate reactor trip.	$\geq$ 15% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% RATED THERMAL POWER. (Full Power Trip)
	10 inadvertent pressurizer auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.

5.0 ADMINISTRATIVE CONTROLS

5.1 RESPONSIBILITY

5.1.1 The Site Director shall be responsible for overall activities of the site, while the Plant Manager shall be responsible for overall operation of the unit. The Site Director and Plant Manager shall each delegate in writing the succession to this responsibility during their absence.

5.1.2 The Shift Operations Supervisor (SOS)(or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Site Director shall be reissued to all station personnel on an annual basis.

5.2 ORGANIZATION

5.2.1 Offsite And Onsite Organizations

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.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the TVA Organizational Topical Report.

b. The Senior Vice President, Nuclear Power shall have corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Site Director shall be responsible for overall activities at the site. The Site Director manages activities associated with site and determines the nature and extent of onsite and offsite services required to support site activities.

d. The Plant Manager shall be responsible for overall unit safe operation, and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.2.1 Offsite And Onsite Organizations (continued)

d. The individuals who carry out radiological control and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Facility Staff

a. Each on duty unit shift shall be composed of at least the minimum shift crew composition shown in Table 5.2-1.

b. At least one licensed Reactor Operator shall be in the unit 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 shall be in the Control Room.

---------------------------NOTE---------------------------
The Radiological Control and Chemistry technician staffing requirements may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.
----

c. A Radiological Control technician and Chemistry technician shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during this operation.

e. A manager who the Shift Operations Supervisor reports to shall hold a Senior Reactor Operator license.

f. Procedures will be established to ensure that NRC policy statement guidelines (Generic Letter No. 82-12) regarding working hours established for employees are followed. In addition, procedures will provide for documentation of authorized deviations from these guidelines and that the documentation is available for NRC review.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

TABLE 5.2-1 (Page 1 of 1)

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SOS	1	1
SRO	1	None
RO	2	1
AUO	4	1
STA	1*	None

SOS - Shift Operations Supervisor with a Senior Reactor Operators License on Unit 1
SRO - Individual with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
AUO - Auxiliary Unit Operator
STA - Shift Technical Advisor

Except for the Shift Operations Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 5.2-1 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 of Table 5.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Operations Supervisor from the Control Room while the unit is in MODES 1, 2, 3 or 4, an individual with a valid SRO license shall be designated to assume the Control Room command function. During an absence of the Shift Operations Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

* The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Operations Supervisor or the individual with a Senior Operator license meet the qualifications for the STA as required by the NRC.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.2.3 Independent Safety Engineering (ISE)

5.2.3.1 Function
The ISE shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources which may indicate areas for improving plant safety.

5.2.3.2 Composition

The ISE shall be composed of at least 3 dedicated full-time engineers located onsite. These engineers will be supplemented as necessary by full-time engineers shared among all TVA nuclear sites to achieve an equivalent staffing of 5 full-time engineers performing the ISE functions applicable to Watts Bar.

5.2.3.3 Responsibilities

----------------------------NOTE--------------------------
ISE members shall not be responsible for sign-off functions.

The ISE shall be responsible for maintaining surveillance of plant activities to provide independent verification that these activities are performed correctly and that human errors are reduced as much as practical.

5.2.3.4 Authority

The ISE shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Manager, Nuclear Manager's Review Group.

5.2.4 Shift Technical Advisor (STA)

5.2.4.1 The STA shall serve in an advisory capacity to the Shift Operations Supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit. The STA shall have a Bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design and transient and accident response and analysis. The STA position may be fulfilled by a person performing collateral duties who fulfills the NRC position requirements.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.3 FACILITY STAFF QUALIFICATIONS

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except those personnel qualifying after January 1, 1990 as the Shift Operations Supervisor (SOS), Assistant Shift Operations Supervisor (ASOS), Reactor Operator (RO), Shift Technical Advisor (STA), and Site Radiological Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, revision 2, April 1987. Personnel qualifying on these positions before January 1, 1990 will still meet the requirements of Regulatory Guide 1.8, revision 1-R May 1977.

5.4 TRAINING

5.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Plant Manager and shall meet or exceed the requirements ANSI N18.1-1971 for comparable positions, except for the SOS, ASOS, and RO whose retraining program shall meet the requirements of 10 CFR 55.59(c) and shall include familiarization with relevant industry operating experience. The STA retraining program shall meet or exceed Regulatory Guide 1.8, revision 2, April 1987. For the SOS, ASOS, RO, STA, and the Site Radiation Protection Manager, the replacement program shall meet the requirements of Regulatory Guide 1.8, revision 2, April 1987.

5.5 REVIEW AND AUDIT

5.5.1 Plant Operations Review Committee (PORC)

5.5.1.1 Function

The PORC shall function to advise the Plant Manager on all matters related to nuclear safety.

5.5.1.2 Composition

The PORC shall be composed of managers or individuals reporting to the managers from the areas listed below:

Member: Plant Manager
Member: Operations
Member: Site Radiological Control
Member: Maintenance
Member: Technical Support
Member: Site Quality Assurance
Member: Nuclear Engineering

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.1.2 Composition (continued)

The Plant Manager shall designate the Chairman in writing from among these members. The chairman and members shall meet the qualification requirements of Specification 5.3.

5.5.1.3 Alternates

All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis. All alternate chairmen and alternate members shall meet the qualification requirements of Specification 5.3.

5.5.1.4 Meeting Frequency

The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

5.5.1.5 Quorum

The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates. No more than two alternates shall participate as voting members in PORC activities at any one time.

5.5.1.6 Responsibilities

The PORC shall be responsible for the activities listed below. The PORC may delegate the performance of reviews to subcommittees or qualified reviewers but will maintain responsibilities for these reviews.

a. Review of: (1) all proposed procedures required by Specification 5.8 and changes thereto, (2) all proposed programs required by Specification 5.9 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications and amendments to the Operating License;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.1.6 Responsibilities (continued)

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Site Director and to the Nuclear Safety Review Board;

f. Review of all Reportable Events;

g. Review of unit operations to detect potential nuclear safety hazards;

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Nuclear Safety Review Board.

i. Review of the Plant Physical Security Plan and implementing procedures;

j. Review of the Site Radiological Emergency Plan and implementing procedures;

K. Review of the Fire Protection Report and implementing procedures:

l. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Site Director and the Nuclear Safety Review Board.

m. Review of changes to the Process Control Program, the Offsite Dose Calculation Manual, and the Radwaste Treatment Systems.

n. Review of changes to the Technical Requirements Manual.

5.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under 5.5.1.6a. through d. prior to their implementation;

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.1.7
(continued)

b. Render determination in writing with regard to whether or not each item considered under 5.5.1.6a. through e. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Site Director and the Nuclear Safety Review Board of disagreement between the PORC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 5.1.1 above.

5.5.1.8 Records

The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Site Director and the Nuclear Safety Review Board.

5.5.1.9 Technical Review And Control

Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Specification 5.8.1 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by a qualified individual other than the individual who prepared the procedure or procedure change, but who may be from the same organization as the individual who prepared the procedure or procedure change. Procedures shall be approved by the appropriate responsible manager as designated in writing by the Plant Manager.

b. Workplans used to implement proposed changes or modifications to structures, systems, and components that affect plant nuclear safety shall be reviewed by a qualified individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Proposed modifications to structures, systems, and components that affect plant nuclear safety and the implementing workplans shall be approved prior to implementation by the appropriate responsible managers as designated in writing by the Plant Manager.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.1.9 Technical Review And Control (continued)

c. Individuals responsible for reviews performed in accordance with Specifications 5.5.1.9a and b, shall be previously designated in writing by the Plant Manager. Each such review shall be performed by qualified personnel of the appropriate discipline and shall include a determination of whether or not additional, cross-disciplinary review is necessary. Each such review shall also include determination of whether or not an unreviewed safety question is involved pursuant to 10 CFR 50.59.

5.5.2 Nuclear Safety Review Board (NSRB)

5.5.2.1 Function

The NSRB shall function to provide for independent review and cognizance of audits to assure adequacy of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

5.5.2.2 Composition

The NSRB shall be composed of at least five members, including the Chairman. Members of the NSRB may be from Nuclear Power, other TVA organizations, or external to TVA.

5.5.2.3 Qualifications

The Chairman, members, and alternate members of the NSRB shall be appointed in writing by the Senior Vice President, Nuclear Power and shall have an academic degree in engineering or a physical science field, or the equivalent; and in addition, shall have a minimum of five years technical experience in one or more areas given in Specification 5.5.2.1. No more than two alternates shall participate as voting members in NSRB activities at any one time.

5.5.2.4 Consultants

Consultants may be utilized as determined by the NSRB Chairman to provide expert advice to the NSRB.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.2.5 Meeting Frequency

The NSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

5.5.2.6 Quorum

The quorum of the NSRB necessary for the performance of the NSRB review and audit functions of these Technical Specifications shall consist of more than half the NSRB membership or at least 5 members, whichever is greater. This quorum shall include the Chairman or his appointed alternate and the NSRB members, including appointed alternate members, meeting the requirements of Specification 5.5.2.3. No more than a minority of the quorum shall have line responsibility for operation of the unit as defined in Specification 5.1.1.

5.5.2.7 Review

The NSRB shall review items b, c, and d below, and shall ensure appropriate review of the remaining subjects to identify any programatic deficiencies:

a. The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes to Technical Specifications or Operating License.

e. Nuclear Safety Significant violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All Reportable Events.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.2.7 Review (continued)

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the PORC.

5.5.2.8 Audits

Audits of unit activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Site Radiological Emergency Plan and implementing procedures at least once per 12 months.

f. The Plant Physical Security Plan, the Safeguards Contingency Plan, and implementing procedures at least once per 12 months.

g. The facility Fire Protection Report and implementing procedures at least once per 24 months.

h. An independent fire protection and loss prevention program inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.5.2.8 Audits (continued)

j. The Radiological Environmental Monitoring Program and the, results thereof at least once per 12 months.

k. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

l. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

m. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

n. The performance of activities required by the Technical Requirements Manual at least once per 24 months.

o. Any other area of unit operation considered appropriate by the NSRB or the Senior Vice President, Nuclear Power.

5.5.2.9 Authority

The NSRB shall report to and advise the Senior Vice President, Nuclear Power on those areas of responsibility specified in Specifications 5.5.2.7 and 5.5.2.8.

5.5.2.10 Records

Records of NSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRB meeting shall be prepared, approved and forwarded to the Senior Vice President, Nuclear Power within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 5.5.2.7 above, shall be prepared, approved and forwarded to the Senior Vice President, Nuclear Power within 14 days following completion of the review.

c. Audit reports encompassed by Specification 5.5.2.8 above, shall be forwarded to the Senior Vice President, Nuclear Power and to the management positions responsible for the areas audited within 30 days after completion of the audit.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.6 REPORTABLE EVENT ACTION

5.6.1 The following actions shall be taken for Reportable Events:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73, and

b. Each Reportable Event shall be reviewed by the PORC and the results of this review shall be submitted to the NSRB and the Site Director.

5.7 SAFETY LIMIT VIOLATION

5.7.1. Safety Limit Violations shall be handled in accordance with the provisions of Specification 2.0 SAFETY LIMITS.

5.8 PROCEDURES

5.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Physical Security Plan implementation.

e. Site Radiological Emergency Plan implementation.

f. Fire Protection Report implementation.

g. Process Control Program implementation.

h. Quality Assurance Program for effluent and environmental monitoring.

i. The emergency operating procedures required to implement the requirements of NUREG-0737 and supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33.

j. Offsite Dose Calculation Manual implementation.

k. Technical Requirements Manual implementation.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.8.2 Each procedure of Specification 5.8.1, and changes thereto, shall be reviewed and approved prior to implementation as set forth in Specification 5.5.1.9 above.

5.8.3 Temporary changes to procedures of Specification 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is approved in accordance with Specification 5.5.1.9 above within 14 days of implementation.

5.9 <u>PROGRAMS</u>

The following programs shall be established, implemented, and maintained.

5.9.1 <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system and waste gas system. 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.

5.9.2 <u>In-Plant Radiation Monitoring</u>

A program which will ensure the capability to accurately determine the airborne iodine concentrations 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.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.3 Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator 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, which shall include monitoring of the condensate for evidence of condenser inleakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for off-control point chemistry conditions,

g. Procedures identifying (a) the authority responsible for the interpretation of the data; and (b) the sequence and timing of administrative events required to initiate corrective action.

5.9.4 Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodides 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,

c. Provisions for maintenance of sampling and analysis equipment.

5.9.5 Radioactive Effluent Controls Program

A program shall be provided conforming with 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 (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.5 Radioactive Effluent Controls Program (continued)

a. Limitations on the operability 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 10 CFR 20, Appendix B, Table II, Column 2,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 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 each unit to Unrestricted Areas conforming to Appendix I to 10 CFR 50,

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 operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50,

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the Site Boundary conforming to the doses associated with 10 CFR 20, Appendix B, Table II, Column 1,

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the Site Boundary conforming to Appendix I to 10 CFR 50,

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.5 Radioactive Effluent Controls Program (continued)

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 greater than 8 days in gaseous effluents released from each unit to areas beyond the Site Boundary conforming to Appendix I to 10 CFR 50,

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.

k. Limitations on the operability of meteorological monitoring instrumentation including surveillance tests in accordance with the methodology in the ODCM.

5.9.6 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

b. A Land Use Census to ensure that changes in the use of areas at and beyond the Site Boundary are identified and that modifications to the monitoring program are made if required by the results of this census, and

c. Participation in a Interlaboratory Comparison program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.7 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations, involving personnel radiation exposure.

5.9.7.1 High Radiation Area

-----------------------NOTE-----------------------
Individuals qualified in radiation protection procedures (e.g. Radiological Control personnel) or personnel escorted by such personnel may be exempt from the Radiation Work Permit (RWP) issuance requirement during the performance of their assigned radiation protection duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they otherwise comply with approved radiation protection procedures for entry into such high radiation areas.
----

a. 100 - 1000 mrem/hr

Pursuant to paragraph 20.203(c)(5) of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a RWP. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by 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.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.7.1 High Radiation Area (continued)

3. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP.

b. > 1,000 mrem/hr

In addition to the requirements of Specification 5.9.7.1.a above, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Operations Supervisor and/or Site Radiological Control Superintendent. Doors shall remain locked except during periods of access by personnel under an approved RWP. 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.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

5.9.8 Process Control Program (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 5.11.2. This documentation shall contain:

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.8 PROCESS CONTROL PROGRAM (PCP) (continued)

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

5.9.9 Offsite Dose Calculation Manual (ODCM)

a. The Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in:

1. The calculation of offsite doses resulting from radioactive gaseous and liquid effluents,

2. The calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and

3. The conduct of the Environmental Radiological Monitoring Program.

b. The ODCM shall also contain:

1. The Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specifications 5.9.5 and 5.9.6, and

2. Descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semiannual Radioactive Effluent Release Reports required by Specifications 5.10.4 and 5.10.5.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.9 Offsite Dose Calculation Manual (ODCM) (continued)

c. Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 5.11.2. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the PORC and the approval of the Plant Manager.

3. 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 Semiannual Radioactive Effluent Release Report for the period of the report in which any change to 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/year) the change was implemented.

5.9.10 Major Changes To Radioactive Waste Treatment Systems

---------------------------NOTE---------------------------
Submittal of information required by this section may be made as part of the annual FSAR update.

Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

a. Shall be reported to the NRC in the Semi-Annual Radioactive Effluent Report for the period in which the evaluation was reviewed in accordance with Specification 5.5.1.9. The discussion of each change shall contain:

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.10 Major Changes To Radioactive Waste Treatment Systems (continued)

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. a detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

4. an evaluation for the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. an evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. a comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. an estimate of the exposure to plant operating personnel as a result of the change; and

8. documentation of the fact that the change was reviewed and found acceptable in accordance with Specification 5.5.1.9.

b. Shall become effective upon review and acceptance in accordance with Specification 5.5.1.9.

5.9.11 Steam Generator Tube Inspection Program

A program for monitoring steam generator tube degradation shall be established. The integrity of each steam generator shall be demonstrated by meeting the requirement of Specification 5.9.14 and by performance of an augmented Inservice Inspection Program which includes at least the following:

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.11 Steam Generator Tube Inspection Program (continued)

a. Steam generator sample selection

b. Steam generator tube sample selection and inspection

   1. Minimum sample size

   2. Inspection result classification

   3. Actions required

c. Inspection frequencies

d. Acceptance Criteria

e. Reports

The Steam Generator Tube Inspection Program and all proposed changes shall be implemented only upon prior approval by the NRC Staff.

5.9.12 Containment Leak Rate Test Program

A program which will ensure that the containment leak rate tests are performed to maintain containment OPERABILITY in accordance with Specification 3.6.1. The program shall include the following surveillances in accordance with 10 CFR 100, Appendix J:

a. Type A test (Overall integrated containment leakage rate).

b. Type B test (Local penetration leak rates).

c. Type C test (Containment isolation valve leakage rates).

d. Air lock seal leakage and air lock overall leakage rates.

5.9.13 Ventilation Systems Filter Testing Program

A program shall be established to implement the following required testing of filters in accordance with [Regulatory Guide 1.52, Rev 2 or ANSI N510-1980].

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.9.13 Ventilation Systems Filter Testing Program (continued)

a. In-place penetration and bypass dioctyl phthalate (DOP) test.

b. In-place penetration and bypass hydrocarbon refrigerant gas test.

c. Methyl iodide penetration test of a charcoal sample.

d. Flow rate and pressure drop test.

e. Heater power test.

5.9.14 Inservice Inspection and Testing Programs

A program shall be established for Inservice Inspection and Testing of ASME Code Class 1, 2, and 3 components. The program shall be reviewed and approved by the Commission and shall, at a minimum:

a. Ensure that Inservice Inspection of ASME Code Class 1, 2, and 3 components and Inservice Testing of ASME Code Class 1, 2, and 3 pumps and valves are 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).

b. Specify the provisions under which relief from the ASME Boiler and Pressure Vessel Code may be sought pursuant to 10 CFR 50, Section 50.55(g)(6)(i).

c. Specify the Surveillance Frequencies for Inservice Inspection and Testing activities.

Any changes to the Inservice Inspection and Testing Programs which could affect the items specified above require prior Commission notification.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10 REPORTING REQUIREMENTS

5.10.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

5.10.2 Startup Reports

a. 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.

b. The Startup Report shall address each of the applicable tests identified in the FSAR and shall 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 requested in license conditions based on other commitments shall be included in this report.

c. Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.3 Annual Reports

-----------------------------NOTE-----------------------------
A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

Annual reports covering the activities of the unit as described below for the previous calendar year shall be sumbitted priot to March 1 of each year. The initial report shall be submitted prior to March 1 fo the year following initial criticality.

-----------------------------NOTE-----------------------------
This tabulation supplements the requirements of 10 CFR 20.407.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis for 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, inservice inspection, routine 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.

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 5.9.11).

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.15. The following information shall be included along with the results of specific activity analysis results in which the primary coolant exceeded the limits of the specifications: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.3 Annual Reports (continued)

limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded: (4) Graph of the I-131 concentration and one other radioiodine isotopic concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

5.10.4 Annual Radiological Environmental Operating Report

----------------------------NOTE----------------------------
A single submittal may be made for a multiple unit station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 30 of each year. The report shall include summaries, interpretations, and an analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

5.10.5 Semiannual Radioactive Effluent Release Report

----------------------------NOTE----------------------------
A single submittal may be made for a multiple unit station.

A Semiannual Radioactive Effluent Release Report Covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR 50.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.6 Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office, no later than the 15th day of each month following the calendar month covered by the report.

5.10.7 Special Reports

Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the following requirements:

a. Moderator Temperature Coefficient (MTC). LCO 3.1.4.

Whenever the MTC is more positive than the upper limit given in Specification 3.1.4, prepare and submit a Special Report to the Commission within [10] days. The report shall describe the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. Accident Monitoring Instrumentation. LCO 3.3.3.

In the event the required number of channels are not restored to OPERABLE status and an alternate method of monitoring is initiated, prepare and submit a Special Report to the commission within the next [14] 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.

c. Fuel Assemblies. Specification 4.3.1.

Whenever more than 10 rods per assembly or 30 rods per core are replaced per refueling with stainless steel filler rods or open water channels for fuel rods, prepare and submit a Special Report to the Commission within 30 days after cycle startup describing the number of rods replaced.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.7 Special Reports (continued)

d. Cold Overpressure Prevention. LCO 3.4.16.

In the event either the Power Operated Relief Valves (PORVs) or the RCS vent(s) are used to mitigate a RCS pressure transient prepare and submit a Special Report to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

e. Emergency Core Cooling Systems (ECCS). LCO 3.5.2 and 3.5.3.

In the event the ECCS is actuated and injects water into the Reactor Coolant system prepare and submit a Special Report to the Commission within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

5.10.8 Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient upper and lower limits and 300 ppm surveillance limit for Specification 3.1.4,
2. Shutdown Bank Insertion Limit for Specification 3.1.6,
3. Control Bank Insertion Limits for Specification 3.1.7,
4. Axial Flux Difference limits, and target band, for Specification 3.2.3,
5. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, F_{xy}, for Specification 3.2.1,
6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier, for Specification 3.2.2.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.8 Core Operating Limits Report (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
   (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.3 - Axial Flux Difference, 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).
   (Methodology for Specification 3.2.3 - Axial Flux Difference (Constant Axial Offset Control).)

3. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
   (Methodology for Specification 3.2.3 - Axial Flux Difference (Constant Axial Offset Control).)

4. NUREG-800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.
   (Methodology for Specification 3.2.3 - Axial Flux Difference (Constant Axial Offset Control).)

5. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.10.8 Core Operating Limits Report (continued)

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided within 30 days of issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

5.10.9 RCS Pressure/Temperature Limits Report

a. RCS pressure and temperature limits including heatup and cooldown rates shall be established and documented in the RCS PRESSURE/TEMPERATURE LIMITS REPORT before the end of the reactor vessel fluence period for Specification 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits including the heatup and cooldown rates shall be those previously reviewed and the approved by the NRC.

c. The RCS pressure and temperature limits including heatup and cooldown rates shall be determined so that all applicable limits (e.g., heatup limits, cooldown limits, inservice leak and hydrostatic testing limits) of the analysis are met.

d. The RCS PRESSURE/TEMPERATURE LIMITS REPORT, including any revisions of supplements thereto shall be provided within 30 days of issuance, for each reactor vessel fluence period, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

5.11 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

5.11.1 5 Year Records

The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.11.1 5 Year Records (continued)

c. All Reportable Events and Special Reports submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 5.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

i. Records of the annual audit of the Radiological Emergency Plan and Implementing Procedures.

j. Records of the annual audit of the Plant Physical Security Plan and Implementing Procedures.

5.11.2 Duration of Operating License

The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 4.7-1.

f. Records of reactor tests and experiments.

(continued)

5.0 ADMINISTRATIVE CONTROLS (continued)

5.11.2 Duration of Operating License (continued)

g. Records of training and qualification for members of the facility staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required for lifetime retention by the Nuclear Quality Assurance Plan.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the NSRB.

l. Records of secondary water sampling and water quality.

m. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This would include procedures effective at specified times and QA records showing that these procedures are followed.

n. Records of reviews performed for changes made to the ODCM and the PCP.

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT
BASES FOR THE
PROPOSED TECHNICAL SPECIFICATIONS

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LIST OF ACRONYMS

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling Water System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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B 2.0 SAFETY LIMITS

BASES

BACKGROUND

The restrictions of Safety Limit 2.1.1 prevent overheating of the fuel and possible cladding failure which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the cladding surface temperature is slightly above the coolant saturation temperature.

The proper functioning of the Reactor Protection System and steam generator safety valves prevent violation of the Safety Limit 2.1.1.

The restrictions of Safety Limit 2.1.2 protect 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, its continued integrity is assured.

The design pressure of the RCS is [2500] psia. As an assurance of system integrity, all RCS components are hydrostatically tested at 125% of design, [3107] psig, per the ASME code requirements prior to initial operation when there is no fuel in the core. Should repairs or replacements take place which would require a full hydrostatic test of the RCS, the fuel would have to be completely off loaded in order to exceed the maximum pressure specified in this LCO. Without fuel in the core there is no chance of fission products getting into the reactor coolant.

Overpressurization of the RCS could result in a breach of the RCS pressure boundary. If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere raising concerns relative to limits on radioactive releases specified in 10 CFR 100.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety Limit 2.1.1

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter, however THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the [W-3 R-grid] correlation (Ref. 1). The [W-3 R-grid] DNB correlation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, 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.

Automatic enforcement of these Reactor Core Safety Limits are provided by the following functions:

a. High Pressurizer Pressure Reactor Trip
b. Low Pressurizer Pressure Reactor Trip
c. Overtemperature Delta-T Reactor Trip
d. Overpower Delta-T Reactor Trip
e. Power Range High Neutron Flux Reactor Trip
f. Steam Generator Safety Valves

General Design Criteria 10 (Ref. 2) requires that the minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients be limited to [1.30]. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The limitation that the average enthalpy in the hot leg be equal to the enthalpy of saturated liquid also ensures that the "Delta-T" measured by instrumentation (used in the protection system design as a measure of the core power) is proportional to core power. It is implicitly assumed in the generation of the core limits and in the operation of the protection functions that the minimum RCS flow requirement of LCO 3.3.1 is satisfied.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Safety Limit 2.1.1 represents a design requirement for establishing the protection system trip setpoints identified above. They are not as restrictive as the conditions of LCO 3.4.1, RCS Pressure, Temperature, and Flow DNB Limits, or the assumed initial conditions of the safety analyses (as indicated in Chapter 15 of the FSAR).

Safety Limit 2.1.2

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10% in accordance with Section III of the ASME Code for Nuclear Power Plant components (Ref. 5). The transient which 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 and nominal feedwater supply is maintained.

More specifically, no credit is taken for operation of the following:

Pressurizer Power-Operated Relief Valves (PORVs),
Steam line relief valve,
Steam Dump System,
Reactor Control System,
Pressurizer Level Control System, or
Pressurizer spray valve.

The RCS pressurizer safety valves, the main steam safety valves, and the reactor high pressure trip have settings established to assure the RCS pressure safety limit will not be exceeded (Ref. 6).

(continued)

BASES (continued)

SAFETY LIMIT

Safety Limit 2.1.1

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and average temperature for which the minimum DNBR is not less than [1.30], or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the [W-3 R-grid] DNBR correlation. The curves are based on enthalpy hot channel factor limits provided in the COLR.

The heat flux Safety Limit is higher than the limit calculated when the AXIAL FLUX DIFFERENCE (AFD) is within the limits of the $f_1(\Delta I)$ function of the Overtemperature Trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits. (Ref. 3, 4)

Safety Limit 2.1.2

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings under [ASME Section III] is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the safety limit on maximum allowable RCS pressure is established at [2735] psig.

APPLICABILITY

Saftey Limit 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 assure operation within the core limits. The steam generator safety valves or automatic protection actions serve to prevent Reactor Coolant System heatup to the core limit conditions or to initiate a reactor trip function (which forces the unit into MODE 3). Setpoints for the reactor trip functions are specified in LCO 3.3.1 and LCO 3.3.2.

Safety Limit 2.1.2 applies in MODES 1 through 5 because it is conceivable to approach or exceed this Safety Limit in these MODES due to overpressurization events. The Safety Limit is not applicable in MODE 6 since the reactor vessel head closure bolts are not fully tightened making it impossible to pressurize the RCS.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATION

2.2.1

When the Safety Limits are exceeded in MODES 1 or 2, the unit must be brought to MODE 3 within 1 hour where the Safety Limit is not applicable. Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling and lead to cladding failure. Similarly, operation with bulk hot leg boiling could limit the effectiveness of the overtemperature/overpower ΔT protection system. A Completion Time of one hour is adequate to accomplish an orderly shutdown.

2.2.2

Exceeding the RCS pressure Safety Limit in MODES 3, 4, or 5 is more severe than in MODES 1 or 2 since the reactor vessel temperature is lower and consequently the vessel material is less ductile. As such, pressure must be reduced to less than the Safety Limit within 5 minutes. 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.

2.2.3 - 2.2.6

Violation of a Safety Limit must be reported to the NRC under 10 CFR 50.73. Operation of the unit is not again permitted until authorized by NRC.

SURVEILLANCE REQUIREMENTS

Although Surveillance Requirements are not applicable to Safety Limits, their intent - assurance of compliance with limits, is satisfied through executing the Surveillance Requirements specified for the Reactor Trip System, pressurizer safety valves, and steam generator safety valves. If maintained OPERABLE, the Reactor Trip System, pressurizer safety valves, and steam generator safety valves will not permit violation of Safety Limits.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section [4.4]

2. Title 10 Code of Federal Regulations (10CFR), Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

3. WCAP 8746-A, Design Bases for the Overtemperature ΔT and the Overpower ΔT Trips, March 1977.

4. WCAP 9273-NP-A, Westinghouse Reload Safety Evaluation Methodology, July 1985.

5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", Article NB-7000, "Protection Against Overpressure", American Society of Mechanical Engineering, [1971].

6. Watts Bar FSAR Chapter [5].

B 3.0 LIMITING CONDITIONS FOR OPERATION (LCO) - APPLICABILITY

BASES

LCOs 3.0.1 - 3.0.4

LCOs 3.0.1 through 3.0.4 establish the general requirements applicable to LCOs. These requirements are based on the requirements for LCOs stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which MODES or other specified conditions) conformance to the LCO is required for safe operation of the unit. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met.

There are two basic types of Required Actions. The first specifies the remedial measures that permit continued operation of the unit which is not further restricted by the time limits of the Required Actions. In this case, conformance to the Required Actions provides an acceptable level of safety for unlimited continued operation as long as the Required Actions continue to be met. The second type of Required Action specifies a time limit in which conformance to the conditions of the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the specified Completion Time, a shutdown is required to place the unit in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown Required Actions be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

(continued)

BASES (continued)

LCO 3.0.1 (continued)

The Completion Times of the Required Actions are applicable from the point in time it is identified that an LCO is not met. The Completion Times of the Required Actions are also applicable when a system or component is removed from service for Surveillance testing or investigation of operational problems.

Individual specifications may include a specified Completion Time of a Surveillance Requirement when equipment is removed from service. In this case, the Completion Times of the Required Actions are applicable when this limit expires if the Surveillance has not been completed.

When a shutdown is required to comply with Required Actions, the unit may have entered a MODE in which a new Specification becomes applicable. In this case, the Completion Times of the Required Actions would apply from the point in time that the new Specification becomes applicable if the requirements of the new LCO are not met. If compliance with the Required Actions of the first Specification is completed in less time than required by the Completion Time, the additional time may be added to the Completion Time of the second Specification.

LCO 3.0.2

LCO 3.0.2 establishes that noncompliance with a specification exists when the requirements of the LCO are not met and the associated Required Actions have not been met within the specified Completion Times. The purpose of this specification is to clarify that: (1) completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification, and (2) completion of the remedial measures of the Required Actions is not required when compliance with an LCO is restored within the Completion Time specified in the associated ACTIONS unless otherwise specified.

LCO 3.0.3

LCO 3.0.3 establishes the shutdown Required Actions that must be implemented when an LCO is not met and the Condition is not specifically addressed by the associated ACTIONS. The purpose of this Specification is to delineate the time limits for placing the unit in a safe shutdown MODE when unit operation cannot be maintained within the limits for safe operation defined by the LCO and its ACTIONS.

(continued)

BASES (continued)

LCO 3.0.3 (continued)

It is not intended to be used as an operational convenience which 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. One hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. 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 cooldown capabilities of the unit assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a unit upset that could challenge safety systems under conditions for which this Specification applies.

If remedial measures permitting limited continued operation of the unit under the provisions of the ACTIONS are completed, the shutdown may be terminated. The Completion Time limits of the ACTIONS are applicable from the point in time it was discovered there was a failure to meet an LCO. Therefore, the shutdown may be terminated if the Required Actions have been met or the Completion Times of the ACTIONS have not expired, thus providing an allowance for the completion of the Required Actions.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation applies. However, if a lower MODE of operation is reached in less time than allowed, 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, the time allowed to reach MODE 4 is the next 11 hours because the total time to reach MODE 4 is not reduced from the allowable limit of 13 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.

The shutdown requirements of LCO 3.0.3 do not apply in MODES 5 and 6, because the ACTIONS of individual specifications define the remedial measures to be taken.

(continued)

BASES (continued)

LCO 3.0.4

LCO 3.0.4 establishes limitations on MODE changes when an LCO is not met. It precludes placing the unit in a different MODE of operation when the requirements for an LCO in the current MODE or the MODE to be entered are not met and continued noncompliance to these conditions would result in a shutdown to comply with the Required Actions if a change in MODES were permitted. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the unit before or after a MODE change. Therefore, in this case, entry into a MODE or other condition specified in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before unit startup.

When a shutdown is required to comply with Required Actions, the provisions of LCO 3.0.4 do not apply because they would delay placing the unit in a lower MODE of operation.

LCO 3.0.5

To be provided

B 3.0 SURVEILLANCE REQUIREMENTS (SR) - APPLICABILITY

BASES

SRs 3.0.1 - 3.0.5	SRs 3.0.1 through 3.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3): "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."
SR 3.0.1	SR 3.0.1 establishes the requirement that Surveillance Requirements must be met during the MODES or other conditions specified in the Applicability for which the requirements of the LCO apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this Specification is to ensure that Surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the unit when the unit is in a MODE or other condition specified in the Applicability for the associated LCO. Surveillance Requirements do not have to be performed when the unit is in a MODE for which the requirements of the associated LCO do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification. Surveillance Requirements do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

(continued)

BASES (continued)

SR 3.0.2

SR 3.0.2 establishes the conditions under which the specified Frequency for Surveillance Requirements, Required Actions which require the performance of a specific Surveillance Requirement, and any Required Action with a Completion Time requiring the periodic performance of an action on a "once per..." interval may be extended. SR 3.0.2 permits an extension of the Frequency to facilitate Surveillance scheduling and consideration of unit operating conditions that may not be suitable for conducting the Surveillance; e.g., transient conditions or other ongoing Surveillance or maintenance activities.

The limit of SR 3.0.2 is based on engineering judgment and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through Surveillance activities is not significantly degraded beyond that obtained from the specified Surveillance Frequency.

SR 3.0.3

SR 3.0.3 establishes the failure to perform a Surveillance within the allowed Surveillance Frequency, defined by the provisions of SR 3.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for an LCO. Under the provisions of this specification, systems and components are assumed to be OPERABLE when the associated Surveillance Requirements have been met. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This Specification also clarifies that the ACTIONS are applicable when Surveillances have not been completed within the allowed Surveillance Frequency and that the Completion Times of the Required Actions apply from the point in time it is identified that a Surveillance has not been performed and not at the time that the allowed Surveillance Frequency was exceeded.

(continued)

BASES (continued)

SR 3.0.3 (continued)

If the Completion Times of the ACTIONS are less than 24 hours or a shutdown is required to comply with Required Actions, e.g., LCO 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the Required Actions. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a Surveillance before a shutdown is required to comply with Required Actions or before other remedial measures would be required that may preclude completion of a Surveillance. The basis for this allowance includes consideration for unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, and the safety significance of the delay in completing the required Surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by Required Actions and for completing Surveillance Requirements that are applicable when an exception to the requirements of SR 3.0.4 is allowed. If a Surveillance is not completed within the 24-hour allowance, the Completion Times of the ACTIONS are applicable at that time.

Completion of the Surveillance Requirement within the Completion Time of the ACTIONS or in accordance with the 24 hour allowance of SR 3.0.3 restores compliance with the requirements of SR 3.0.1. However, this does not negate the fact that the failure to have performed the Surveillance within the Surveillance Frequency, defined by the provisions of SR 3.0.2, was a violation of the OPERABILITY requirements of an LCO that is subject to enforcement action. Further, the failure to perform a Surveillance within the provisions of SR 3.0.2 is a violation of a Technical Specification requirement and is, therefore, a Reportable Event under the requirements of 10 CFR 50.73(a)(2)(i)(8) because it is a condition prohibited by the unit's Technical Specifications.

(continued)

BASES (continued)

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable Surveillances must be met before entry into a MODE or other condition specified in the Applicability. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition specified in the Applicability for which these systems and components ensure safe operation of the unit. This provision applies to changes in MODES or other conditions specified in the Applicability associated with unit shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be met to ensure that the LCO is met during initial unit startup or following a unit outage.

When a shutdown is required to comply with Required Actions, the provisions of SR 3.0.4 do not apply because this would delay placing the unit in a lower MODE of operation.

B 3.1 REACTIVITY CONTROLS

B 3.1.1 Shutdown Margin - T_{avg} > 200 °F

BASES

BACKGROUND

SHUTDOWN MARGIN requirements provide sufficient reactivity margin to assure the core will remain subcritical following all transients and design basis events. As such, the SHUTDOWN MARGIN defines the amount of negative reactivity which must be present immediately following the insertion of all shutdown and control rods, assuming the single rod cluster assembly of highest reactivity worth is fully withdrawn.

General Design Criteria 26 (Ref. 1) requires that two independent reactivity control systems be provided, and that one of these systems be capable of holding the core subcritical under cold conditions. These requirements are met by the use of movable control assemblies and soluble boric acid in the reactor coolant system. 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 provides the minimum shutdown margin under Condition I events 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 boron system can compensate for all xenon burnout reactivity changes and will maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system that satisfies GDC 26.

During power operation, SHUTDOWN MARGIN control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.5, Control Bank Insertion Limits. When in the shutdown MODES, the SHUTDOWN MARGIN requirements are met by adjustments to the Reactor Coolant System (RCS) boron concentration.

APPLICABLE SAFETY ANALYSES

The required SHUTDOWN MARGIN is assumed as an initial condition in safety analysis. Specifically, the primary safety analyses which rely on the SHUTDOWN MARGIN limits are the Steamline Break Analysis and the Boron Dilution Analysis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

For the Steamline Break Analysis, the SHUTDOWN MARGIN requirement is the major factor limiting the power level which the core may reach during a return to criticality. This event is most limiting at end of core life when the moderator temperature coefficient is most negative.

In the Boron Dilution Analysis, the required SHUTDOWN MARGIN defines the difference between an initial boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the reactor coolant system and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The assumption of a shutdown margin may be utilized in analyses which are primarily evaluated following reactor trip such as Feedline Break, Loss of Normal Feedwater, and Steamline Break Mass/Energy Release analyses. The cases where this is utilized, however, are unit specific.

Other design basis accidents, such as Rod Withdrawal during startup, do not assume a specific value for SHUTDOWN MARGIN, but rather assume only that sufficient negative reactivity to remain subcritical is provided by the insertion of all rods, except the rod of highest reactivity worth, which is assumed to remain fully withdrawn.

SHUTDOWN MARGIN satisfies the requirements of Selection Criterion 2 (Ref. 2). Even though it is not directly observed from the control room, SHUTDOWN MARGIN is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCOs

The Steam Line Break accident and the Boron Dilution accidents are generally the most limiting analyses which explicitly use SHUTDOWN MARGIN. For Steam Line Break accidents, if the LCO is violated there is a potential to exceed the Departure from Nucleate Boiling Ratio and to

(continued)

BASES (continued)

LCOs (continued)	exceed 10 CFR 100 limits. The Steam Line Break mass and energy release results could also be adversely affected if the LCO is violated. For the Boron Dilution accident, if the LCO is violated, then the minimum required time for operator action to terminate dilution may no longer be available.
APPLICABILITY	SHUTDOWN MARGIN requirements for MODES 1 and 2 with K_{eff} > 1.0 are the bank insertion limits of LCO 3.1.6, Shutdown Bank Insertion Limit, and LCO 3.1.7, Control Bank Insertion Limits. These limits provide assurance that SHUTDOWN MARGIN requirements are being met. The required SHUTDOWN MARGIN is applicable for MODE 2 with k_{eff} < 1.0, and MODES 3 and 4. This assures that the transients analyzed for these MODES have sufficient reactivity as discussed above. The shutdown reactivity requirements for MODE 6 are given in LCO 3.9.1, Boron Concentration.
ACTIONS	<u>A.1</u> If the SHUTDOWN MARGIN requirements are not met, boration must be initiated within 15 minutes. In the determination of the required combination of boration flow rate and boron concentration, there is no unique design basis event which must be satisfied. The only requirement is to restore the SHUTDOWN MARGIN to its required value as soon as possible since the SHUTDOWN MARGIN constitutes an initial condition in many safety analyses. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boration solution should be a highly concentrated solution, such as that normally found in the boric acid tanks. In determining the boration flow rate, the following points should be considered: a. The most difficult time in core life to increase the RCS boron concentration is at Beginning of Cycle (BOC) when the boron concentration may approach or exceed 2000 ppm.

(continued)

BASES (continued)

ACTIONS (continued)

A.1 (continued)

b. Although the amount of SHUTDOWN MARGIN which must be recovered is variable, for the example below, assume a value of 1% Δk/k must be recovered.

Using the above assumptions 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 SHUTDOWN MARGIN by 1% Δk/k. These boration parameters of [10] gpm and [20,000] ppm represent typical values and are provided for the purpose of offering a specific example. In general, the action timing and flow rates will provide for prompt restoration of the SHUTDOWN MARGIN.

The above discussion does not account for reactivity changes due to xenon decay or RCS cooldown because it is not required to do so. Immediately after a trip, the xenon concentration in the core begins increasing, effectively adding 1% to 3% Δk/k negative reactivity approximately 8 hours after shutdown. It is not until 20 to 30 hours after shutdown that xenon decay begins to add net positive reactivity to the reactor.

RCS cooldown in going from MODE 2 with k_{eff} < 1.0 or MODE 3 to MODE 5 results in positive reactivity additions. This MODE change normally occurs relatively slowly over several hours.

The effects of xenon occur on a relatively long time scale as do the effects of changes in the RCS temperature associated with normal MODE changes. The above boron concentrations and boration rates will recover lost SHUTDOWN MARGIN and easily compensate for xenon decay and MODE changes.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SHUTDOWN MARGIN is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration
b. Control bank position
c. Shutdown Bank Position
d. RCS average temperature
e. Fuel burnup based on gross thermal energy generation
f. Xenon concentration
g. Samarium concentration
h. Isothermal Temperature Coefficient (ITC)

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical in the applicable MODES for this specification and the fuel temperature will be changing at the same rate as the RCS. The surveillance frequency of 24 hours allows time enough to collect the required data, including a boron concentration analysis and complete the calculation, and yet is often enough to maintain surveillance on a parameter which normally varies slowly.

SR 3.1.1.2

When a rod, or group of rods, is known to be immovable or untrippable, there is a potential to impact the SHUTDOWN MARGIN. In this situation, it is important to promptly verify that the SHUTDOWN MARGIN requirements are met. Since the core conditions and the SHUTDOWN MARGIN may change with time, it is also important to periodically verify that the SHUTDOWN MARGIN requirements continue to be met.

For these reasons the surveillance frequencies are specified as within 1 hour and 12 hours. If the rod will not trip, or cannot be moved due to mechanical interference or excessive friction, it cannot be assumed to be fully inserted when assessing the actual SHUTDOWN MARGIN.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations (10 CFR), Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commissions' Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

B 3.1 REACTIVITY CONTROLS

B 3.1.2 Shutdown Margin - $T_{avg} \leq 200$ °F

BASES

BACKGROUND

SHUTDOWN MARGIN requirements provide sufficient reactivity margin to assure the core will remain subcritical following all transients and design basis events. As such, the SHUTDOWN MARGIN defines the amount of negative reactivity which must be present immediately following the insertion of all shutdown and control rods, assuming the single rod cluster assembly of highest reactivity worth is fully withdrawn.

General Design Criteria 26 (Ref. 1) requires that two independent reactivity control systems be provided, and that one of these systems be capable of holding the core subcritical under cold conditions. These requirements are met by the use of movable control assemblies and soluble boric acid in the reactor coolant system. 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 provides the minimum shutdown margin under Condition I events 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 boron system can compensate for all xenon burnout reactivity changes and will maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system that satisfies GDC 26.

During power operation, SHUTDOWN MARGIN control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.5, Control Bank Insertion Limits. When in the shutdown MODES, the SHUTDOWN MARGIN requirements are met by adjustments to the Reactor Coolant System (RCS) boron concentration.

APPLICABLE SAFETY ANALYSES

The required SHUTDOWN MARGIN is assumed as an initial condition in safety analysis. Specifically, the primary safety analysis which rely on the SHUTDOWN MARGIN limits in MODE 5 is the Boron Dilution Analysis.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

In the Boron Dilution Analysis, the required SHUTDOWN MARGIN defines the difference between an initial boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the reactor coolant system and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The assumption of a shutdown margin may be utilized in analyses which are primarily evaluated following reactor trip such as Feedline Break, Loss of Normal Feedwater, and Steamline Break Mass/Energy Release analyses. The cases where this is utilized, however, are unit specific..

Other design basis accidents, such as Rod Withdrawal during startup, do not assume a specific value for SHUTDOWN MARGIN, but rather assume only that sufficient negative reactivity to remain subcritical is provided by the insertion of all rods, except the rod of highest reactivity worth, which is assumed to remain fully withdrawn.

SHUTDOWN MARGIN satisfies the requirements of Selection Criterion 2 (Ref. 2). Even though it is not directly observed from the control room, SHUTDOWN MARGIN is considered an initial condition process variable because it is periodically monitored to provide assurance that the unit is operating within the bounds of accident analysis assumptions.

LCOs

The Boron Dilution accident is generally the most limiting MODE 5 analysis which explicitly used SHUTDOWN MARGIN. For the Boron Dilution accident, if the LCO is violated, then the minimum required time for operator action to terminate dilution may no longer be available.

APPLICABILITY

The required SHUTDOWN MARGIN is applicable for MODE 5. This assures that the transients analyzed for this MODE have sufficient reactivity as discussed above. The shutdown reactivity requirements for MODE 6 are given in LCO 3.9.1, Boron Concentration.

(continued)

BASES (continued)

ACTIONS

A.1

If the SHUTDOWN MARGIN requirements are not met, boration must be initiated within 15 minutes.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique design basis event which must be satisfied. The only requirement is to restore the SHUTDOWN MARGIN to its required value as soon as possible since the SHUTDOWN MARGIN constitutes an initial condition in many safety analyses.

Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boration solution should be a highly concentrated solution, such as that normally found in the boric acid tanks.

In determining the boration flow rate, the following points should be considered:

a. The most difficult time in core life to increase the RCS boron concentration is at Beginning of Cycle (BOC) when the boron concentration may approach or exceed 2000 ppm.

b. Although the amount of SHUTDOWN MARGIN which must be recovered is variable, for the example below, assume a value of 1% $\Delta k/k$ must be recovered.

Using the above assumptions 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 SHUTDOWN MARGIN by 1% $\Delta k/k$. These boration parameters of [10] gpm and [20,000] ppm represent typical values and are provided for the purpose of offering a specific example. In general, the action timing and flow rates will provide for prompt restoration of the SHUTDOWN MARGIN.

The above discussion does not account for reactivity changes due to xenon decay or RCS cooldown because it is not required to do so. Immediately after a trip, the xenon concentration in the core begins increasing, effectively adding 1% to 3% $\Delta k/k$ negative reactivity approximately 8 hours after shutdown. It is not until 20 to 30 hours after shutdown that xenon decay begins to add net positive reactivity to the reactor.

(continued)

BASES (continued)

ACTIONS (continued)

RCS cooldown in going from MODE 2 with $k_{eff} < 1.0$ or MODE 3 to MODE 5 results in positive reactivity additions. This MODE change normally occurs relatively slowly over several hours.

The effects of xenon occur on a relatively long time scale as do the effects of changes in the RCS temperature associated with normal MODE changes. The above boron concentrations and boration rates will recover lost SHUTDOWN MARGIN and easily compensate for xenon decay and MODE changes.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The SHUTDOWN MARGIN is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration
b. Control bank position
c. Shutdown Bank Position
d. RCS average temperature
e. Fuel burnup based on gross thermal energy generation
f. Xenon concentration
g. Samarium concentration
h. Isothermal Temperature Coefficient (ITC)

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical in the applicable MODES for this specification and the fuel temperature will be changing at the same rate as the RCS. The surveillance frequency of 24 hours allows time enough to collect the required data, including a boron concentration analysis and complete the calculation, and yet is often enough to maintain surveillance on a parameter which normally varies slowly.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.2.2

When a rod, or group of rods, is known to be immovable or untrippable, there is a potential to impact the SHUTDOWN MARGIN. In this situation, it is important to promptly verify that the SHUTDOWN MARGIN requirements are met. Since the core conditions and the SHUTDOWN MARGIN may change with time, it is also important to periodically verify that the SHUTDOWN MARGIN requirements continue to be met.
For these reasons the surveillance frequencies are specified as within 1 hour and 12 hours. If the rod will not trip, or cannot be moved due to mechanical interference or excessive friction, it cannot be assumed to be fully inserted when assessing the actual SHUTDOWN MARGIN.

REFERENCES

1. Title 10 Code of Federal Regulations (10 CFR), Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commissions' Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

B 3.1 REACTIVITY CONTROLS

B 3.1.3 Core Reactivity

BASES

BACKGROUND

The limitation on core reactivity is provided to ensure that the value reflects the limiting conditions assumed in the FSAR accident and transient analyses.

When overall core reactivity varies from the prediction by as much as 1% Δk/k, the core design should be assessed to identify the cause of the deviation if possible, but at least determine that continued operation will remain within the bounds of the safety analyses.

APPLICABLE SAFETY ANALYSES

The safety analyses are evaluated for each core design. The applicable safety analyses includes all accidents and transients which include assumed values of reactivity coefficients, control and shutdown rod worths, and power distributions. (Ref. 1)

Overall core reactivity satisfies Selection Criterion 2 of the NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (Ref. 2), since it monitors core reactivity which is an initial condition assumed in accident analyses.

LCOs

The LCO requires that the overall core reactivity reflect that calculated in the core design. If the actual overall reactivity deviates significantly from the prediction, one of the many components that cause reactivity to vary may be in error. If the core design contains such an error, safety parameters calculated or evaluated in the safety analyses are in question.

If overall core reactivity varies from the prediction with burnup, it could also indicate an anomalous physical phenomenon occurring in the core region which would not have been considered in the safety analyses thereby invalidating the safety analyses.

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ because the core must be critical to assess the overall reactivity.

ACTIONS

A.1

The core design assessment should include the following parameters:

- Doppler temperature coefficient
- Doppler power coefficient
- Prompt neutron lifetime
- Trip reactivity from full power
- Shutdown margin at hot zero power
- K_{eff} at hot zero power with two stuck rods

72 hours to complete the core design assessment is acceptable given the limited magnitude of the deviation, which should not invalidate the safety analyses.

A.2

The establishment of appropriate operating restrictions and surveillance requirements will ensure that operation would continue within the bounds of the reanalysis. Experience has shown that a completion time of 72 hours to institute the new requirements is adequate.

B.1

If the Required Action and associated Completion Time is not met, the unit must be placed in MODE 3 in 6 hours. This allows adequate time to shut the unit down without challenging safety systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

Comparing the overall core reactivity balance to the predicted values each 31 EFPD is sufficient to observe a reactivity change which could approach the LCO limit. Generally, a deviation in core reactivity will vary slowly with burnup.

The core reactivity balance should consider the following factors:

- Reactor Coolant System boron concentration,
- Control Rod position,
- Reactor Coolant System average temperature,
- Fuel burnup based on gross thermal energy generation,
- Xenon concentration, and
- Samarium concentration.

After refueling, the core parameters will be measured and compared to design predictions to confirm the results of the core design. Overall reactivity will be confirmed as consistent with core design safety parameters during reload startup physics testing. Adjustment, which could be normalization of the all-rods-out equilibrium boron concentration, of predicted reactivity to actual reactivity allows subsequent surveillances to monitor the changes which occur with cycle burnup. A burnup of 60 EFPD is adequate to obtain equilibrium conditions after startup physics testing to assess the overall core reactivity.

REFERENCES

1. WCAP 9272, Westinghouse Reload Safety Evaluation Methodology, March 1978.

2. 52FR3788, Proposed Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

B 3.1 REACTIVITY CONTROL

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of unit conditions. Accordingly, verification of MTC values measured at conditions other than those explicitly stated will require extrapolation to those specified conditions in order to permit an accurate comparison.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria which prohibit a return to criticality or the DNBR criteria of the approved correlation which could lead to a loss of the fuel cladding integrity.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE SAFETY ANALYSES

The Final Safety Analysis Report (FSAR) contains analyses of accidents that result in both overheating and overcooling of the reactor core. The Moderator Temperature Coefficient is a controlling parameter in most of these accidents. Therefore, both the maximum positive value and maximum negative value of the MTC are important to safety, and both values must be bounded. Values used in the FSAR consider worst case conditions to ensure the accident results are bounding.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero or RATED THERMAL POWER (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. Such accidents include sudden feedwater flow increase, sudden decrease in feedwater temperature, and steamline break events.

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 the beginning or end of cycle life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 1).

MTC satisfies the requirements of Selection Criterion 2 (Ref. 2), even though it is not directly observed and controlled from the control room. MTC is considered an initial condition process variable because it is periodically monitored to provide operators with information that assures the unit is operating within the bounds of the accident analysis assumptions.

LCOs

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 Beginning of Cycle (BOC); this upper bound must not be exceeded. At End of Cycle (EOC) the MTC takes on its most negative value, he the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

The NOTE establishes a maximum positive value that can not be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

(continued)

BASES (continued)

APPLICABILITY

Technical Specifications place both LCO and Surveillance Requirements values on MTC, based on the safety analysis assumptions described above. The most positive MTC LCO limit applies to MODES 1 and 2 with $k_{eff} > 1.0$, and requires that the MTC be less positive than the specified limit value. The most negative MTC LCO limit applies to MODES 1, 2, and 3 and requires that the MTC be less negative than the specified limit value for all rods withdrawn, end of cycle life, RATED THERMAL POWER condition. Maintaining the MTC within these limits ensures that it remains consistent with design basis assumptions.

ACTIONS

A.1

If the upper limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time to evaluate the MTC measurement and compute 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 BOC MTC will meet the LCO requirement can be determined. At this point in core life, Condition A no longer exists, the unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the rods are inadvertently placed in a condition that is not within the withdrawal limits, the operator must immediately restore the rods to within the withdrawal limits.

C.1

If the required administrative withdrawal limits at BOC are not established in 24 hours, the unit must be placed in MODE 2 with $K_{eff} < 1.0$ to prevent operation with an MTC which is more positive than that assumed in safety analyses. The Completion Time of 6 hours provides sufficient time for an orderly shutdown.

(continued)

BASES

ACTIONS (continued)

D.1

Exceeding the lower MTC limit means that the safety analysis assumptions for the EOC accidents which use a bounding negative MTC value may be invalid. If the lower MTC limit is exceeded, the unit must be in MODE 4 in 12 hours. A Completion Time of 12 hours allows for an orderly shutdown.

SURVEILLANCE REQUIREMENTS

The note states that SR 3.0.4 is not applicable since the unit must be in MODES 1 or 2 to measure the MTC.

SR 3.1.4.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the limit prior to entering MODE 1 assures that the limit will also be met at higher power levels.

The upper MTC value for All Rods Out (ARO) will be inferred from Isothermal Temperature Coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the upper MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.4.2 and SR 3.1.4.3

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm Surveillance Requirement value of MTC should necessarily be less negative than the lower MTC limit. The 300 ppm Surveillance Requirement value is sufficiently less negative than the lower MTC limit value to provide assurance that the LCO limit will be met when the 300 ppm surveillance criterion is met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

If the 300 ppm surveillance limit is exceeded, it is possible that the lower limit on MTC could be reached before the planned end of the cycle. Because the MTC changes slowly with core depletion, the surveillance frequency of 14 Effective Full Power Days (EFPD) is sufficient to avoid exceeding the lower limit.

The surveillance limit for RTP boron concentration of 60 ppm is conservative in the following way. If the measured MTC at 60 ppm is less negative than the 60 ppm surveillance limit the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. WCAP 9273-NP-A, Westinghouse Reload Safety Evaluation Methodology, July 1985.

2. T.E. Murley to W.S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

B 3.1 REACTIVITY CONTROL

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up/withdrawn) or into the core (down/inserted) by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inches) at a time but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be 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. Unit 1 has control banks and at least 4 shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. 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 position of maximum withdrawal 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 approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. 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, which are the Bank Demand Position Indication system (commonly called group step counters), and the Analog Rod Position Indication (ARPI) 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

(continued)

BASES

BACKGROUND (continued)	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 accurate (± 1 step or ± 5/8 inch). 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 ARPI system provides a highly reliable indication of actual control rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center-to-center distance of 3.75 inches which is 6 steps. The normal indication accuracy of the ARPI system is ± 3.5 steps (± 2.2 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 15.5 steps, or 10 inches.
	The applicable General Design Criteria for the movable control assemblies and their position indication systems are GDC 26 and GDC 28 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The operability of the shutdown and control rods are initial assumptions in all safety analyses which assume rod insertion upon reactor trip. Maximum rod misalignment directly affects core power distributions and assumptions of available SHUTDOWN MARGIN.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two types of analysis are performed in regard to static rod misalignment. With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on DNBR in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another 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 SHUTDOWN MARGIN is met with the maximum worth RCCA also fully withdrawn.

Shutdown and control rod operability and alignment are directly related to power distributions and SHUTDOWN MARGINS, which are initial conditions assumed in safety analyses. Therefore they may be considered process variables that satisfy the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

The limits on shutdown or control rod alignments assure that the assumptions in the safety analysis will remain valid. The requirements on operability assure that upon reactor trip, the assumed reactivity will be available and will be inserted. The operability requirements also assure that the RCCAs and banks will move correctly upon command, to maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

BASES

LCOs (continued)

Failure to meet the requirements of this LCO may produce power distributions with unacceptable peaking factors and linear heating rates, or unacceptable SHUTDOWN MARGINs, all of which may constitute initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only modes in which power is generated, and the operability and alignment of rods has the potential to affect the safety of the plant. In the shutdown modes, the OPERABILITY of the shutdown and control rods has the potential to affect the required SHUTDOWN MARGIN, but this effect can be compensated for by an increase in the boron concentration of the reactor coolant system.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable to the extent that they are immovable and untrippable there is a high probability that the required SHUTDOWN MARGIN may be adversely affected. Under these conditions it is important to determine the SHUTDOWN MARGIN, and if it is less than the required value, initiate boration until the required SHUTDOWN MARGIN is recovered. The Completion Time of 1 hour is adequate to determine SHUTDOWN MARGIN and, if necessary, to initiate boration to restore SHUTDOWN MARGIN.

A.2

In addition to the actions required in A.1.1 and A.1.2.1, the unit must be put in MODE 3 since the accident analysis assumption of only one stuck rod is no longer valid. The Completion Time of 6 hours allows the operator sufficient time to perform an orderly shutdown of the reactor.

(continued)

BASES

ACTIONS (continued)

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant and operation may proceed without further restriction.

B.2

An alternative to realigning a single misaligned RCCA to the group average position is aligning the remainder of the group to the position of the misaligned or inoperable RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, Shutdown Bank Insertion Limit, and LCO 3.1.7, Control Bank Insertion Limits. The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.3.1.1 and B.3.1.2

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning Control Bank B to a rod that is misaligned fifteen steps from the top of the core would require a significant power reduction since Control Bank D must be moved fully in and Control Bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but inoperable or misaligned provided that SHUTDOWN MARGIN is verified within 1 hour or boration is initiated within 1 hour to establish the required SHUTDOWN MARGIN. The Completion Time of 1 hour represents the time necessary to determine the actual unit SHUTDOWN MARGIN and, if necessary align and start the necessary systems and components to initiate boration.

(continued)

BASES

ACTIONS (continued)

B.3.2, B.3.3, B.3.4, B.3.5, B.3.6, and B.3.7

Reduction of power to 75% of RATED THERMAL POWER (RTP) ensures that local linear heat rate increases due to a misaligned RCCA will not cause the core design criteria to be exceeded. The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the reactor protection system.

Reduction of the Power Range Neutron Flux--High trip setpoints to 85% of RTP after power has been reduced to 75% of RTP maintains both core protection and an operability margin at reduced power similar to that at full power. The Completion Time of 6 hours (4 hours after power has been reduced) allows sufficient time to plan, schedule, and complete the adjustments of the overpower trip setpoint.

When a rod, or group of rods, is known to be inoperable or misaligned, there is a potential to impact the SHUTDOWN MARGIN. Since the core conditions can change with time, periodic verification of SHUTDOWN MARGIN is required. A surveillance frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that Heat Flux Hot Channel Factor - $F_Q(Z)$ and Nuclear Enthalpy Hot Channel Factor - $F_{\Delta H}{}^N$ are within the required limits ensures that current operation at 75% of RTP with a rod misaligned is not resulting in power distributions which could cause fuel damage if the reactor were 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 for calculation of $F_Q(Z)$ 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 event for the duration of operation under these conditions. The list of Safety Analyses requiring a reevaluation are continued in Table B 3.1.5-1. 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

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing may continue to cause misalignment, the unit must be shutdown and the rods restored to an operable status. No deliberate attempts to restore the rods to within alignment limits should be made. The Completion Time of 6 hours allows sufficient time to shutdown the unit in a controlled manner.

D.1

In most cases, when more than one rod is found to be trippable and aligned but inoperable, the malfunction can be traced to the Rod Control System. Since the majority of Rod Control System malfunctions can be repaired without reactor shutdown and since the unit conditions are not outside any accident analysis assumptions, there is time available to locate the malfunction and restore the rods to an OPERABLE status. Maintaining the sequence, insertion, and power limits of LCO 3.1.6 (Shutdown Bank Insertion Limits) and LCO 3.1.7 (Control Bank Insertion Limits) ensures that core design limits are not exceeded. A Completion Time of 72 hours provides adequate time for location of the malfunction as well as obtaining parts and performing the repairs.

E.1

When Required Actions B or D cannot be completed within their Completion Time, the unit must be placed in MODE 3 within 6 hours. This LCO is not applicable in MODE 3; placing the unit in MODE 3 alleviates concerns about the development of undesirable xenon or power distributions. The Completion Time of 6 hours is reasonable, based upon operating experience, to reach MODE 3 without challenging the safety systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual rod positions are within the alignment limits at a frequency of 12 hours provides a history that allows the operator to detect a rod beginning to deviate from its expected position. If the Rod Position Deviation Monitor is inoperable, a frequency of 4 hours accomplishes the same goal.

SR 3.1.5.2

Exercising control rod groups at a frequency of 31 days allows the operator to determine that all rods continue to be OPERABLE, even if they are not regularly moved.

SR 3.1.5.3

Individual rods whose drop times are greater than safety analysis assumptions are not OPERABLE. 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 assures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time. During normal operation, performing the tests at a frequency of 18 months assures no degradation in these systems has occurred that would adversely affect control rod motion or drop time.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

Table B 3.1.5-1 (Page 1 of 1)

Safety Analyses Requiring Reevaluation
in the Event of an Inoperable Rod

SAFETY ANALYSES
1. Rod Cluster Control Assembly Insertion Characteristics
2. Rod Cluster Control Assembly Misalignment
3. Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System
4. Single Rod Cluster Control Assembly Withdrawal at Full Power
5. Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)
6. Major Secondary Coolant System Pipe Rupture
7. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

B 3.1 REACTIVITY CONTROLS

B 3.1.6 Shutdown Bank Insertion Limit

BASES

BACKGROUND

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up/withdrawn) or into the core (down/inserted) by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inches) at a time but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be 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. Unit 1 has control banks and at least 4 shutdown banks.

The RCCAs are one of the two independent core reactivity control systems required by General Design Criteria (GDC) 26 (Ref. 1). The total number of RCCAs, the arrangement of control banks and shutdown banks, and the design of the control system satisfy the requirements of GDC 27, GDC 28, and GDC 29 (Ref. 1).

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). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal operations. Hence they are not capable of adding a large amount of positive reactivity. Boration or dilution of the reactor coolant system compensates for the reactivity changes associated with large changes in reactor coolant system temperature.

(continued)

BASES

BACKGROUND (continued)

The shutdown banks are used primarily to help ensure that the required SHUTDOWN MARGIN is maintained. The shutdown banks are controlled manually by the control room operator. During normal unit operation the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shutdown. They add negative reactivity to shutdown the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core (Ref. 1). The shutdown banks shall be within their insertion limits 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 3.1.5, Control Bank Insertion Limits. The Shutdown Bank Insertion Limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SHUTDOWN MARGIN following a reactor trip from full power. The combination of control banks and shutdown banks (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 SHUTDOWN MARGIN at rated no-load temperature (Ref. 2). The shutdown bank insertion limit also limits the reactivity worth of an ejected Shutdown Rod.

The Shutdown Bank Insertion Limit preserves an initial condition assumed in the safety analyses and, as such, satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCOs

The shutdown banks must be within their insertion limits anytime the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SHUTDOWN MARGIN following a reactor trip.

The shutdown bank insertion limits are defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY

The Shutdown Banks must be within their insertion limits with the reactor in MODE 1,and in MODE 2 when k_{eff} is ≥ 1.0, and within 15 minutes prior to initial control bank withdrawal during an approach to criticality. This ensures that a sufficient amount of negative reactivity is available to shutdown the reactor and maintain the required SHUTDOWN MARGIN following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. The reactor is not critical or approaching criticality in MODES 4, 5, or 6 and therefore, the shutdown banks need not be within their insertion limits.

The requirements of this LCO are not applicable during the performance of Surveillance Requirement 3.1.3.2 which requires that the RCCAs be moved at least every 31 days to verify their OPERABILITY. The individual RCCAs are moved at least 10 steps and then returned a position within their physical insertion limit.

ACTIONS

<u>A.1</u>

When one or more shutdown banks is not within insertion limits, 2 hours are allowed to restore the shutdown banks to the within the insertion limits. This is necessary because the available SHUTDOWN MARGIN is significantly reduced with one or more of the shutdown banks not within their insertion limits. The Completion Time of 2 hours is acceptable based on the time requirements for evaluating and repairing minor problems and the low probability of an accident occurring within this short period of time.

(continued)

BASES

ACTIONS (continued)

B.1

If the Shutdown Banks cannot be within their insertion limits within 2 hours, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. Required Action places the unit in MODE 2 with $k_{eff} < 1.0$ within 8 hours. The Completion Time of 6 hours is based on industry operating experience and will not place unnecessary stress on the unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that the shutdown banks are within their insertion limits within 15 minutes prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shutdown the reactor and the required SHUTDOWN MARGIN is maintained following a reactor trip. This Surveillance Requirement and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are only positioned manually by the control room operator, a verification of shutdown bank position at a frequency of 12 hours after the reactor is taken critical is adequate to ensure that they are within their insertion limits.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. Watts Bar FSAR, Section [4.3].

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

B 3.1 REACTIVITY CONTROLS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up/withdrawn) or into the core (down/inserted) by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inches) at a time but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be 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. Unit 1 has control banks and at least 4 shutdown banks.

The RCCAs are one of the two independent core reactivity control systems required by General Design Criteria (GDC) 26 (Ref. 1). The total number of RCCAs, the arrangement of control banks and shutdown banks, and the design of the control system satisfy the requirements of GDC 27, 28, and 29 (Ref. 1).

The control bank insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR). An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance travelled together by two control banks. Therefore, for a 113 step overlap of Figure B 3.1.7-1, Control Banks A and B will travel 113 steps together. The predetermined position of Control Bank C at which Control Bank D will begin to move with Bank C on a withdrawal will be at 118 steps for a fully withdrawn position of 231 steps. The fully withdrawn position is defined in the COLR.

(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 reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited so that the fuel design criteria (Ref. 1 and 2) are maintained. Together, LCO 3.1.6, Shutdown Bank Insertion Limit, LCO 3.1.7, 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 & control bank insertion limits, AFD, and QPTR are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the control bank insertion limits limit the ejected rod worth and the shutdown and control bank insertion limits assure the required SHUTDOWN MARGIN 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 LOCA, loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The Bank Insertion Limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by a RTS trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SHUTDOWN MARGIN requirement is assured by limiting the allowable inserted worth of the RCCAs so that sufficient reactivity is available in the rods to shutdown the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip.

Operation at the insertion limits and/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 which have sufficiently high ejected RCCA worths.

The Bank Insertion Limits ensure that safety analyses assumptions for SHUTDOWN MARGIN, ejected rod worth, and power distribution peaking factors are preserved. The Insertion Limits are a process variable that is an initial condition of a Design Basis Accident or Transient Analysis that presents a challenge to the primary fission product barrier and, as such, satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

The limits on control banks sequence, overlap and physical insertion specified in the COLR must be maintained because they serve the dual function of preserving power distribution and assuring that the SHUTDOWN MARGIN is maintained.

These limits are determined so as to maintain acceptable power distributions during normal operation and acceptable consequences following a postulated rod ejection accident. 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.

(continued)

BASES (continued)

APPLICABILITY

The control banks sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODE 1, and in MODE 2 when k_{eff} is $\geq$ 1.0. These limits must be maintained since they preserve the power distribution, ejected rod worth, and assure meeting SHUTDOWN MARGIN assumptions. Applicability in MODE 2 with k_{eff} < 1.0, and MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. Additionally, SHUTDOWN MARGIN is explicitly required in these MODES by LCO 3.1.1 and LCO 3.1.2.

The requirements of this LCO are not applicable during the performance of SR 3.1.5.2 which requires that the RCCAs be moved at least every 31 days to verify their OPERABILITY. The individual RCCAs are moved at least 10 steps and then returned to their original position. This exemption exists because it is not the point of surveillance requirements to cause LCO violations. Power distribution, SHUTDOWN MARGIN, and ejected rod worth assumptions remain intact during the performance of this surveillance.

ACTIONS

A.1 and B.1

When the control banks are below the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways: 1) reducing power to consistent with rod position, or 2) moving rods to be consistent with power.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

(continued)

BASES

ACTIONS

A.1 and B.1 (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, or ejected rod accident has an acceptably low probability. The Completion Time of 2 hours is acceptable based on the time requirements for evaluating and repairing minor problems and the low probability of an accident occurring within this short period of time.

C.1

When Required Actions A.1 or B.1 cannot be completed within their required Completion Time, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. The LCO is not applicable in MODE 2 with $k_{eff} < 1.0$ since neither the power distribution nor reactivity criteria would be exceeded in this MODE. The Completion Time of 6 hours is based upon unit operating experience. This Completion Time will not place unnecessary stress on the unit systems or operators to reach MODE 2 with $k_{eff} < 1.0$ from full power conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits. Limits on control bank insertion while the reactor is critical also assure that required SHUTDOWN MARGIN assumptions are maintained.

The Estimated Critical Position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP were calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden -- there are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control banks 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 control bank insertion limits. Should the insertion limit monitor become inoperable, verification of the control banks position at a frequency of 4 hours is sufficient to detect control banks that may be approaching the control banks insertion limits.

Surveillance Requirement 3.0.4 is not applicable since the unit must be in the applicable MODES in order to perform surveillances which demonstrate the LCO limits are met.

SR 3.1.7.3

When control banks are maintained within their insertion limits, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the CORE OPERATING LIMITS REPORT. A surveillance frequency of 12 hours is sufficient to ensure compliance with these requirements.

Surveillance Requirement 3.0.4 is not applicable since the unit must be in the applicable MODES in order to perform surveillances which demonstrate the LCO limits are met.

REFERENCES

1. Title 10 Code of Federal Regulations (10 CFR), Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

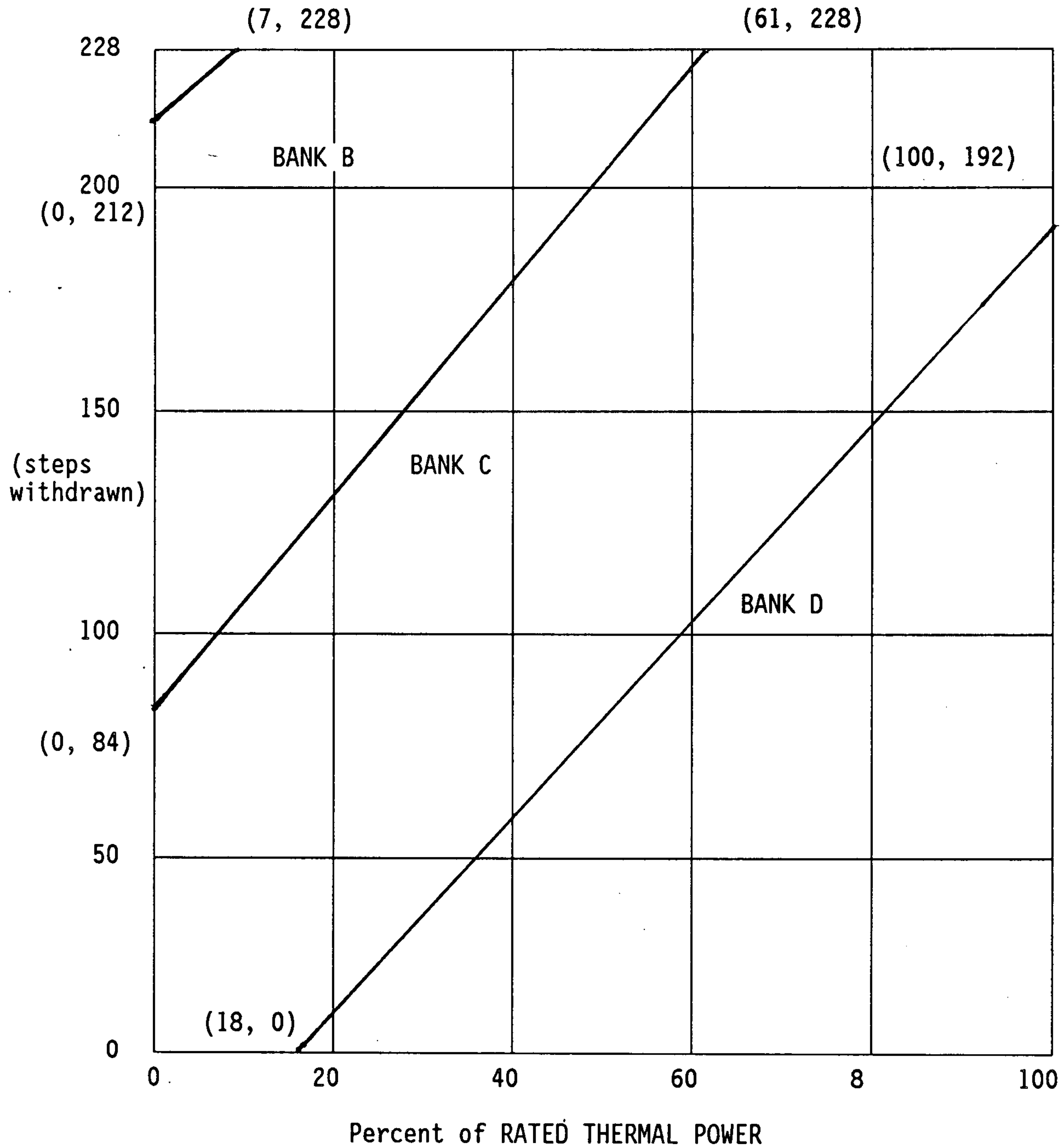


Figure B 3.1.5-1 (Page 1 of 1)

Control Bank Insertion Limits vs Percent RATED THERMAL POWER

B 3.1 REACTIVITY CONTROLS

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

Rod Cluster Control Assemblies (RCCAs), or rods, are moved out of the core (up/withdrawn) or into the core (down/inserted) by their Control Rod Drive Mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inches) at a time but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be 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. Unit 1 has control banks and at least 4 shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. 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 position of maximum withdrawal 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 approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. 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 are determined by two separate and independent systems, which are the Bank Demand Position Indication system (commonly called group step counters), and the Analog Rod Position Indication (ARPI) system.

(continued)

BASES (continued)

BACKGROUND (continued)

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 accurate (± 1 step or ± 5/8 inch). 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 ARPI system provides a highly reliable indication of actual control rod position, but at a lower accuracy than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center-to-center distance of 3.75 inches, which is 6 steps. The normal indication accuracy of the ARPI system is ± 3.5 steps (±2.2 inches). With an indicated deviation of 12 steps between the group step counter and ARPI, the maximum deviation between actual rod position and the demand position could be 15.5 steps, or 10 inches.

The applicable General Design Criteria (GDC) for the movable control assemblies and their position indication systems are GDC 26 and GDC 28 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SHUTDOWN MARGINS, which are initial conditions assumed in safety analyses. Therefore, they may be considered process variables that satisfy the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 2). Although the LCO on Rod Position Indication does not satisfy the Selection Criteria, it is retained as a means to verify that operability and alignment requirements for shutdown and control rods are met.

LCOs

OPERABILITY of the rod position indicators is required to determine rod positions and thereby ensures compliance with the rod alignment and bank insertion limits requirements of LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, and LCO 3.1.7, Control Bank Insertion Limits.

(continued)

BASES (continued)

APPLICABILITY

The requirements on the ARPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, 3.1.6, and 3.1.7) because these are the only modes in which power is generated, and the operability and alignment of rods has 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 SHUTDOWN MARGIN, but this effect can be compensated for by an increase in the boron concentration of the reactor coolant system.

ACTIONS

A.1

When one or more analog rod position indicator channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Since normal power operation does not require excessive movement of banks, verification of RCCA position within the Completion Time of 8 hours is adequate to allow continued full power operation.

A.2

Reduction of THERMAL POWER to ≤ 50% of RATED THERMAL POWER (RTP) within the Completion Time of 8 hours puts the core into a condition where rod position isn't significantly affecting core peaking factors. This Completion Time is based upon engineering judgement and the low probability of a design basis accident within this period of time.

B.1 and B.2

When one or more rods with inoperable position indicators has been moved in excess of 24 steps in one direction since its position was last determined, actions must be promptly initiated to begin verifying that these rods are still properly positioned relative to their group positions. The Completion Time of 15 minutes is long enough to allow preparation of the movable incore detector system and short enough to prohibit any unnecessary delays in doing so.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2 (continued)

If within 8 hours the rod positions have not been determined, THERMAL POWER must be < 50% of RTP to avoid undesirable power distributions that could result from continued operation above 50% of RTP if one or more rods are misaligned by more than 24 steps. This required Completion Time of 8 hours also sets the Completion Time of 8 hours for completing the determination of rod positions.

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the analog rod position indication system. Since normal power operation does not require excessive movement of rods, verification that the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the Completion Time of once every 8 hours is adequate to allow continued full power operation.

C.2

Reduction of THERMAL POWER to ≤ 50% of RTP within the Completion Time of 8 hours puts the core into a condition where rod position isn't significantly affecting core peaking factors. This Completion Time is based upon engineering judgement and the low probability of a design basis accident within this period of time.

D.1

When Required Actions cannot be completed within their required Completion Time, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. This LCO is not applicable in MODE 2 since neither the power distribution nor reactivity criteria would be exceeded in this MODE. The Completion Time of 6 hours is based upon industry operating experience. This Completion Time will not place unnecessary stress on the unit systems or operators to reach MODE 3 from full power conditions.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that individual rod positions are within the group average height limits at the frequency of SR 3.1.5.1 provides a history that allows the operator to detect a rod beginning to deviate from its expected position.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. 52FR3788, "Interim Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

B 3.1 REACTIVITY CONTROLS

B 3.1.9 Mode 1 Physics Tests Exceptions

BASES

BACKGROUND

The primary purpose of the MODE 1 Physics Tests Exceptions is to permit relaxations of existing LCOs to allow the performance of instrumentation calibration tests and special PHYSICS TESTS if a need should arise to do so. The exceptions to LCO 3.2.3, Axial Flux Difference, and LCO 3.2.4, Quadrant Power Tilt Ratio, are most often required for instrumentation calibration tests at the beginning of each cycle. The exceptions to LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, and LCO 3.1.7, Control Bank Insertion Limits, may be required in the event that it is necessary or desireable to do special PHYSICS TESTS involving abnormal rod or bank configurations.

Section XI of 10 CFR Part 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 power unit as specified in General Design Criterion (GDC) 1, Quality Standards and Records (Ref. 2).

The key objectives of a test program are: to provide assurance that the facility has been adequately designed; to validate the analytical models used in the design and analysis; to verify the assumptions used to predict unit response; to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design; and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high powers, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles assure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 3).

(continued)

BASES

BACKGROUND

The PHYSICS TESTS required for reload fuel cycles (Ref. 3) are listed below. None of these tests should result in violating the LCOs to which exception may be taken by this LCO.

a. Critical Boron Concentration - Control Rods Withdrawn,

b. Critical Boron Concentration - Control Rods Inserted,

c. Control Rod Group Worth,

d. Isothermal Temperature Coefficient,

e. Neutron Flux Symmetry,

f. Power Distribution - Intermediate Power,

g. Power Distribution - Full Power.

The first four tests are performed in MODE 2, the next test can be performed in MODES 1 or 2, and the last two tests are performed in MODE 1. 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 an example, the AXIAL FLUX DIFFERENCE calibration test may require very deep control bank insertions. The last two tests are performed at ≥ 90% of RTP.

The Neutron Flux Symmetry Test measures the degree of azimuthal symmetry of the core neutron flux at as low a power level as practical, depending on the method used. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% of RTP.

(continued)

BASES

BACKGROUND (continued)

The Power Distribution - Intermediate Power Test measures the power distribution of the reactor core at intermediate power levels between 40 and 75% of RTP. This test uses the incore flux detectors to measure core power distribution.

The Power Distribution - Full Power Test measures the power distribution of the reactor core at ≥ 90% of RTP using the incore flux detectors.

For initial startups there are two currently required tests which violate the referenced LCOs. The pseudo-ejected rod test performed at approximately 30% of RTP and the pseudo-dropped rod test performed at approximately 50% of RTP require individual rod misalignments which exceed the limits specified in relevant LCOs.

In the event additional special tests may be necessary at some point after the initial startup testing program, this LCO is retained.

APPLICABLE SAFETY ANALYSIS

The fuel is protected by Technical Specification LCOs which preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs which are superseded by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 4) and are specified for each fuel cycle in the CORE OPERATING LIMITS REPORT. 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 controls or process variables to deviate from their LCO limitations. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, LCO 3.1.7, Control Bank Insertion Limits, LCO 3.2, Axial Flux Difference, or LCO 3.2.4, Quadrant Power Tilt Ratio, are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, Heat Flux Hot Channel Factor, and LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor are satisfied. Therefore, LCO 3.1.7 requires surveillance of the hot channel factors to verify that their limits are not being exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Although the PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables that satisfy the requirements of Selection Criteria 1, 2, and 3 of the NRC Interim Policy Statement (Ref. 5). Among the process variables involved are Axial Flux Difference and Quadrant Power Tilt Ratio which represent initial conditions of the unit Safety Analyses. Also involved are the movable control components (control and shutdown rods) which are required to shutdown the reactor.

The MODE 1 Physics Tests Exceptions were not evaluated against the Selection Criteria (Ref. 6). Reference 6 allows the Physics Tests Exceptions to be included as part of the LCOs which they affect. It was decided to retain the Physics Tests Exceptions as separate LCOs because this approach was less cumbersome than including the requirements in the affected LCOs.

LCOs

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1 to verify certain core physics parameters. The power level is limited to ≤ 85% of RTP and the Power Range Neutron Flux trip setpoint is set at 10% of RTP above the PHYSICS TESTS power level with a maximum setting of 90% of RTP. Violation of LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, LCO 3.1.7, Control Bank Insertion Limits, LCO 3.2.3, Axial Flux Difference, or LCO 3.2.4, Quadrant Power Tilt Ratio, during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the requirements of LCO 3.2.1, Heat Flux Hot Channel Factor, and LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor are satisfied.

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 1 when performing PHYSICS TESTS. The applicable PHYSICS TESTS are performed at ≤ 85% of RTP. Other PHYSICS TESTS are performed at full power but do not require violation of any existing LCOs and therefore do not require a PHYSICS TESTS exception. The PHYSICS TESTS performed in MODE 2 are covered by LCO 3.1.10, MODE 2 PHYSICS TESTS exceptions.

ACTIONS

A.1 and A.2

When THERMAL POWER is > 85% of RTP, the only acceptable actions are to reduce THERMAL POWER to ≤ 85% of RTP or to suspend the PHYSICS TESTS exceptions. Fuel integrity may be challenged with control rods or shutdown rods misaligned and THERMAL POWER above 85% of RTP. The Completion Time of 1 hour is based on engineering judgment and the practical amount of time it may take to perform the Required Action. This Completion Time is consistent with the Required Actions of the LCOs suspended by the PHYSICS TESTS.

B.1 and B.2

When the Power Range Neutron Flux--High trip setpoints are > 10% of RTP above the PHYSICS TESTS power level or > 90% of RTP, the Reactor Trip System cannot provide the required degree of core protection if the trip setpoint is greater than the specified value.

The only acceptable actions are to restore the trip setpoint to the allowed value or to suspend PHYSICS TESTS exceptions. The Completion Time of 1 hour is based on engineering judgement and the practical amount of time it may take to perform the Required Actions. This Completion Time is consistent with the Required Actions of the LCOs suspended by the PHYSICS TESTS.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the power level is ≤ 85% of RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A surveillance frequency of 1 hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.1.9.2

Verification of the Power Range Neutron Flux--High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the Reactor Trip System (RTS) is properly set to perform PHYSICS TESTS. Verifying the trip setpoints at a frequency of 8 hours during the performance of the PHYSICS TESTS ensures that the RTS will provide the required core protection.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core Heat Flux Hot Channel Factor and the Nuclear Enthalpy Rise Hot Channel Factor. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits while other LCO requirements are suspended. The frequency of 12 hours is based on engineering judgement and the practical amount of time that it may take to run an incore flux map and calculate the hot channel factors.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, 1988.

2. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

3. ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors, American National Standards Institute, December 13, 1985.

4. WCAP-9273-NP-A, Westinghouse Reload Safety Evaluation Methodology Report, July 1985.

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

6. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

B 3.1 REACTIVITY CONTROLS

B 3.1.10 Mode 2 Physics Tests Exceptions

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 Part 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 power unit as specified in General Design Criterion (GDC) 1, Quality Standards and Records (Ref. 2).

The key objectives of a test program are: to provide assurance that the facility has been adequately designed; to validate the analytical models used in the design and analysis; to verify the assumptions used to predict unit response; to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design; and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high powers, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles assure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 3).

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables that satisfy the Selection Criteria Requirements (Ref. 4 and 5). Among the process variables involved are Axial Flux Difference and Quadrant Power Tilt Ratio which represent initial conditions of the unit Safety Analyses. Also involved are the movable control components (control and shutdown rods) which are required to shutdown the reactor. The PHYSICS TESTS required for reload fuel

(continued)

BASES

BACKGROUND (continued)

cycles (Ref. 3) are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn,

b. Critical Boron Concentration - Control Rods Inserted,

c. Control Rod Group Worth,

d. Isothermal Temperature Coefficient,

e. Neutron Flux Symmetry,

f. Power Distribution - Intermediate Power,

g. Power Distribution - Full Power.

The first four tests are performed in MODE 2, the next test can be performed in MODES 1 or 2, and the last two tests are performed in MODE 1. 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.

The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at Hot Zero Power (HZP) with all rods out, the lead control bank is at or near its fully withdrawn position. Hot Zero Power is a reactor operating state where the core is essentially critical ($k_{eff} = 1.0$), but is not producing measurable heat from nuclear fission, the reactivity due to xenon is negligible, and the RCS is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP with a control bank having a worth of at least 1% when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all control banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected control bank is then inserted to make up for the decreasing boron concentration until the

(continued)

BASES

BACKGROUND (continued)

selected control bank has been moved over its entire range of travel. The reactivity resulting from each incremental control bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the control bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured control bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, Rod Group Alignment Limits, 3.1.6, Shutdown Bank Insertion Limit, or LCO 3.1.7, Control Bank Insertion Limits.

The Control Rod Group Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has 2 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 Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is determined by measuring the reactivity added by withdrawal of the reference bank. This sequence is repeated as necessary for the remaining control banks. This sequence is repeated for the remaining control banks. Performance of this test could violate LCOs 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, or LCO 3.1.7, Control Bank Insertion Limit.

(continued)

BASES

BACKGROUND (continued)

The Isothermal Temperature Coefficient Test (ITC) measures the ITC of the reactor. This test is performed at HZP and has 2 methods of performance. The first method, the Slope Method, varies Reactor Coolant System (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 vs. 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 second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is merely the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, Minimum Temperature for Criticality.

The Neutron Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% of RATED THERMAL POWER (RTP)(Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, or LCO 3.1.7, Control Bank Insertion Limits. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% of RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by Technical Specification LCOs which preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs which are exempted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 6) and are specified for each fuel cycle in the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

CORE OPERATING LIMITS REPORT. 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 controls or process variables to deviate from their LCO limitations. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, Moderator Temperature Coefficient, LCO 3.1.5, Rod Group Alignment Limits, LCO 3.1.6, Shutdown Bank Insertion Limit, LCO 3.1.7, Control Bank Insertion Limits, and LCO 3.3.2, Minimum Temperature for Criticality, are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% of RTP and the reactor coolant temperature is kept ≥ [541]°F.

The MODE 2 Physics Tests Exceptions were not evaluated against the Selection Criteria (Ref. 4 and 5). Reference 5 allows the Physics Tests Exceptions to be included as part of the LCOs which they affect. It was decided to retain the Physics Tests Exceptions as separate LCOs because this approach was less cumbersome than including the requirements in the affected LCOs.

LCOs

This LCO allows the reactor parameters of moderator temperature coefficient and minimum temperature for criticality to be outside their specified limits. In addition, it also allows selected control and shutdown rods to be positioned outside of their specified alignment limits and insertion limits. The THERMAL POWER is limited to less than or equal to 5% of RTP and the RCS temperature is kept greater than or equal to [541]°F. These transgressions beyond specified limits are permitted for the purpose of performing PHYSICS TESTS and pose no threat to fuel integrity provided the Surveillance Requirements are met.

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. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, MODE 1 Physics Tests Exceptions.

(continued)

BASES (continued)

ACTIONS

A.1

When THERMAL POWER is > 5% of RTP, the only acceptable action is to open the reactor trip breakers to prevent operation of the reactor beyond its design limits. Immediately opening the reactor trip breakers will shutdown the reactor and prevent operation of the reactor outside of its design limits.

B.1 and B.2

When RCS lowest T_{avg} is < [541]°F, the only acceptable actions are to restore T_{avg} to within its specified limit within 15 minutes or place the unit in MODE 3 within the next 15 minutes. The LCO is no longer applicable in MODE 3. The Completion Time of 15 minutes is based on engineering judgment and the practical amount of time that it may take to perform the Required Action. Operation with the reactor critical and with temperature below [541]°F could violate the assumptions for the steam break accident and other accidents analyzed in the safety analyses. The additional 15 minutes to place the unit in MODE 3 is based on engineering judgment and unit operating experience, in order not to place unnecessary stress on unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The Power Range and Intermediate Range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1. An ANALOG CHANNEL OPERATIONAL TEST is performed within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the Reactor Trip System is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.10.2

Verification that the RCS lowest T_{avg} is $\geq$ [541]°F will ensure that the unit is not operating in a condition which could invalidate the safety analyses. Verification of the RCS temperature at a frequency of 30 minutes during the performance of the PHYSICS TESTS will provide assurance that the initial conditions of the safety analyses are not violated.

SR 3.1.10.3

Verification that the power level is $\leq$ 5% of RTP will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Unit operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level at a frequency of 1 hour is sufficient to ensure that the THERMAL POWER does not exceed the limit.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

2. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants.

3. ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors, American National Standards Institute, December 13, 1985.

4. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

5. 52FR3788, "Interim Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

6. WCAP-9273-NP-A, Westinghouse Reload Safety Evaluation Methodology, July 1985.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$ (F_{XY} Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of Heat Flux Hot Channel Factor, $F_Q(Z)$, is to limit the local (pellet) peak power density.

The 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 fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, Axial Flux Difference (AFD), and LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), which are directly and continuously measurable process variables. These LCOs along with LCO 3.1.7, Control Bank Insertion Limits, preserve core limits on a continuous basis.

$F_Q(Z)$ changes with 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; its limits are preserved by this LCO. Measurements are generally taken with the core at or near steady-state conditions. Core monitoring and control under non-steady-state conditions is accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and control rod insertion limits.

APPLICABLE SAFETY ANALYSES

Limits on $F_Q(Z)$ preserve the initial total peaking factor assumed in the accident analyses. During large and small break Loss of Coolant Accident (LOCA), the peak cladding temperature must not exceed a limit of 2200°F (Ref. 1). Other criteria must be met - maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling, but peak cladding temperature is typically most limiting. The fuel is protected in part by Technical Specification LCOs that preserve safety and accident initial conditions criteria on power peaking.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting (lower) relative to the F_Q assumed in safety analyses for other accidents. The LCO provides, therefore, conservative limits for other accidents. Power peaking is an initial condition assumed in safety analyses as such. $F_Q(Z)$ satisfies Selection Criterion 2 (Ref. 2).
LCOs	The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a large and small break LOCA. $F_Q(Z)$ is limited by the relationships which are contained in the CORE OPERATING LIMITS REPORT (COLR). This LCO requires operation within the bounds assumed in the safety analysis. Calculations are performed in the core design process to confirm that the core can be controlled during operation to meet the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, core power reduction is required. Violating the LCO on $F_Q(Z)$ could produce unacceptable consequences should a design basis event occur while $F_Q(Z)$ is outside the limits.
APPLICABILITY	The $F_Q(Z)$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the limits assumed in the accident analyses. Applicability in other MODES is not required because there is insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power. The value of $F_Q(Z)$ in these MODES is therefore not important.

(continued)

BASES (continued)

ACTIONS

A.1

Reducing THERMAL POWER by 1% for each 1% $F_Q(Z)$ exceeds its limit maintains an acceptable absolute power density. A Completion Time of 15 minutes is acceptable to perform the tradeoff with THERMAL POWER given the low probability of a LOCA during this time.

A.2

When core peaking factors are sufficiently high that LCO 3.2.1 does not permit operation at 100% of RTP, the Acceptable Operation Limits for AXIAL FLUX DIFFERENCE are scaled down as specified in the COLR. This assures a near constant maximum linear heat rate (Kw/ft) at the Acceptable Operation Limits. The Completion Time of 4 hours for the setpoints change is sufficient considering the small likelihood of a transient in this period and the preceding reduction in THERMAL POWER.

A.3

Reduction in the Power Range Neutron Flux--High trip setpoints is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 8 hours for the setpoints change is sufficient considering the small likelihood of a transient in this period and the preceding reduction in THERMAL POWER.

A.4

Reduction in the Overpower ΔT Trip setpoints is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a transient in this period and the preceding reduction in THERMAL POWER.

A.5

Verification that $F_Q(Z)$ has been restored to within its limit is a conservative action to prevent any deliberate power increase with $F_Q(Z)$ outside its limit. In order to assure that core conditions during operation at higher power levels are consistent with safety analyses assumptions, the value of $F_Q(Z)$ must be verified as being within limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

The nuclear design includes calculations which predict that the core can be operated within the $F_Q(Z)$ limits. Since flux maps are taken at steady state conditions, the axial variations in power distribution for normal operation maneuvers are not present in the flux map data. These axial variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. $F_{XY}(Z)$ is the fundamental radial peaking factor which is a component of $F_Q(Z)$ and should be consistent between the nuclear design values and the measured values ($F_{XY}(Z)$ multiplied by the normalized average axial power at elevation Z gives $F_Q(Z)$).

The following terms are used in the F_{XY} evaluation:

F_{XY}^M The measured value of F_{XY} obtained directly from the flux map results

F_{XY}^C The measured value, F_{XY}^M multiplied by [1.08] to account for fuel manufacturing tolerances and flux map measurement uncertainty.

F_{XY}^{RTP} The limit of F_{XY} at RTP.

$F_{XY}L$ The limit of F_{XY} at the current THERMAL POWER level.

F_{XY}^{RTP} and the F_{XY}^L equation are provided in the COLR. F_{XY}^M and F_{XY}^C are understood to be measured and calculated at discrete core elevations. For example, F_{XY} implies $F_{XY}(Z)$. Flux map data will typically be taken for 30-75 core elevations.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

1. The Core plane regions applicable to an F_{XY} evaluation exclude the following, measured in percent of core height:

   a. Lower core region, from 0 to 15% inclusive,

   b. Upper core region, from 85 to 100% inclusive,

   c. Grid plane regions, ± 2% of the bank demand position of the control banks.

The top and bottom regions of the core are excluded from the F_{XY} evaluation due to the difficulty of making a precise and meaningful measurement in this region and also because of the low probability that this region would be more limiting than the central 70% of the core in the accident analyses. Grid plane regions and rod tip regions are excluded because the flux data may give spurious values due to the difficulty in lining up flux traces accurately in regions of rapidly varying flux. In addition, these are reduced in local power density from neutron absorption in the grids and control rods and cannot be regions of peak linear power.

An evaluation of $F_{XY}(Z)$ is used to confirm that $F_Q(Z)$ is within its limits. If $F_{XY}{}^C$ is less than $F_{XY}{}^{RTP}$, it is concluded that the LCO limit on $F_Q(Z)$ is met. This is true for flux maps taken at reduced power since the $F_{XY}(Z)$ value will inherently be less as THERMAL POWER is increased. Doppler and moderator feedback will flatten the power distribution with increased THERMAL POWER.

If $F_{XY}{}^C$ at less than RTP is greater than $F_{XY}{}^{RTP}$ but less than $F_{XY}{}^L$, SR 3.2.1.2 is repeated to demonstrate that $F_{XY}(Z)$ is being sufficiently reduced as power increases so that when RTP is attained the measured $F_{XY}(Z)$ will be less than $F_{XY}{}^{RTP}$.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

If THERMAL POWER has been increased by 20% or more of RTP since the last determination of $F_{XY}{}^C$, another measurement and F_{XY} evaluation is required within 24 hours to assure that F_{XY} values are being reduced sufficiently to meet the LCO limits with power increases.

If $F_{XY}{}^C$ at less than RTP is greater than $F_{XY}{}^L$, the $F_Q(Z)$ limit may be exceeded. As specified in the COLR, proportionally increasing the predicted $F_Q(Z)$ by the amount that $F_{XY}{}^L$ is exceeded gives an adjusted $F_Q(Z)$ which is compared to the $F_Q(Z)$ limit. If the adjusted $F_Q(Z)$ exceeds the LCO limit, Condition A is entered.

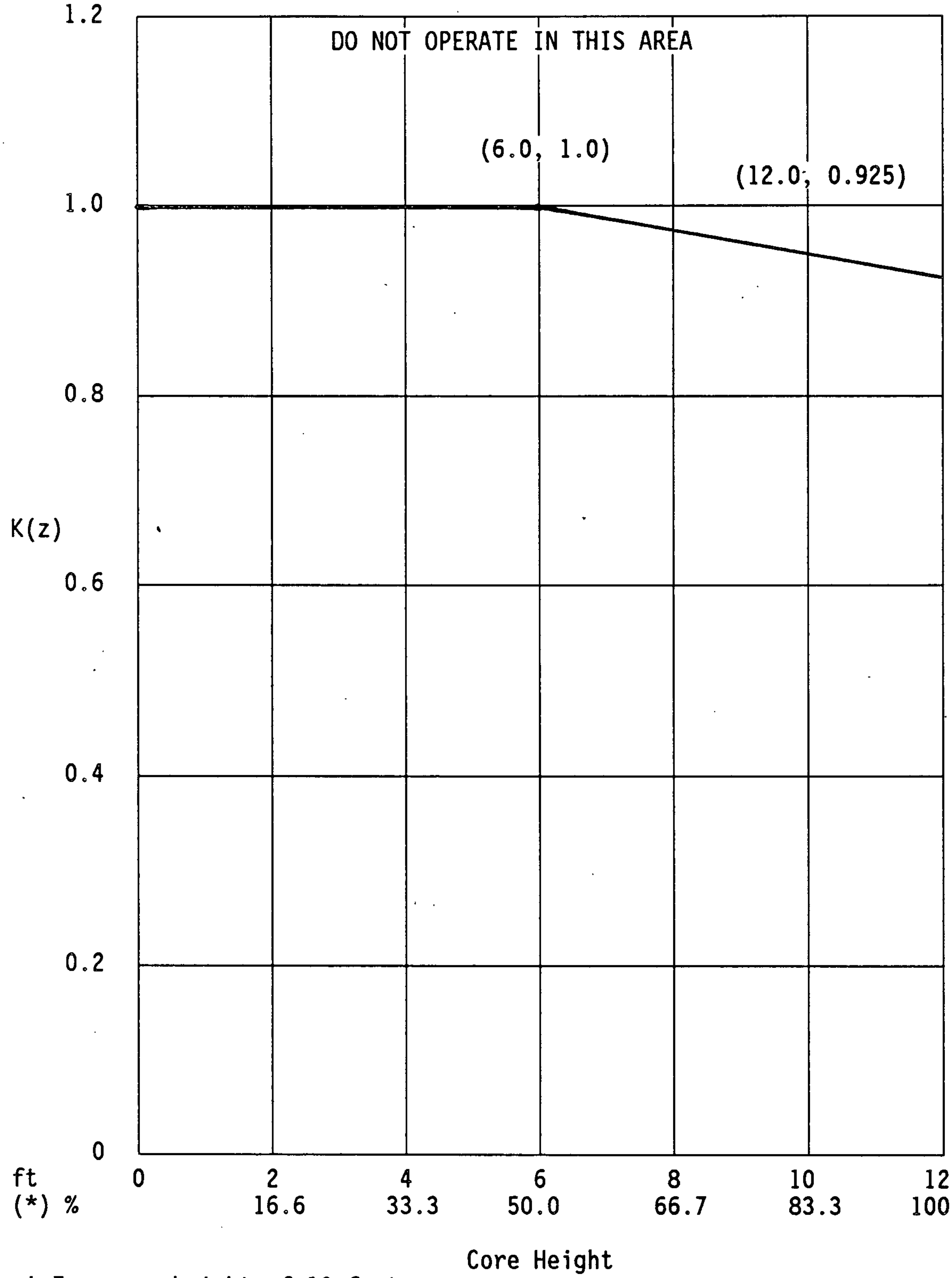
The surveillance frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The surveillance may be done more frequently if required by the results of F_{XY} evaluations. The F_{XY} evaluation is required by SR 3.2.1.2 if the evaluation shows that $F_{XY}{}^{RTP}$ is < $F_{XY}{}^C$.

Performing the surveillance prior to exceeding 75% of RATED THERMAL POWER assures that the $F_Q(Z)$ limit will be met when RATED THERMAL POWER is achieved.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR Part 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.

2. T. E. Murley to W. S. Wilgus, NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications, May 9, 1988 (NRC letter).



* For core height of 12 feet

Figure B 3.2.1-1 (Page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 REACTOR

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor-$F_{\Delta H}^N$

BASES

BACKGROUND

The power density at any point in the core must be limited so that the fuel design criteria (Ref. 1, and 2) and accident analysis assumptions are not exceeded. 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 within these factors assures local conditions in the fuel rods and coolant channels will not challenge core integrity at any location during normal operation or an accident as analyzed in the safety analyses.

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$, 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.

$F_{\Delta H}^N$ is sensitive to fuel loading patterns, bank insertion, flow, 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 by using the movable incore detector system. $F_{\Delta H}^N$ is calculated at least every 31 Effective Full Power Days (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.

(continued)

BASES

BACKGROUND (continued)

The CORE OPERATING LIMIT REPORT (COLR) provides peaking factor limits which ensure that the Departure from Nucleate Boiling (DNB) design basis is met for normal operation, operational transients, and any transient condition arising from faults of moderate frequency. The DNB design basis precludes departure from nucleate boiling and is met by limiting the minimum DNBR (local DNB heat flux ratio) to [1.3]. All transient events that may be DNB limited are assumed to begin with a $F_{\Delta H}^N$ which satisfies the LCO requirements.

Operation outside the LCO limits could produce unacceptable consequences should a DNB limiting event occur. The DNB design basis ensures that there will be no overheating of the fuel which may result in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel must not sustain damage as a result of normal operation and anticipated operational occurrences (Condition I and II, Ref. 3). Limits on $F_{\Delta H}^N$ prevent core power distributions from occurring which would exceed the following fuel design limits:

1. There must be at least a 95% probability at a 95% confidence level (the 95/95 limit) that the hottest fuel rod in the core does not experience a DNB condition (Ref. 6).

2. During a large break Loss Of Coolant Accident (LOCA), the Peak Cladding Temperature (PCT) must not exceed a limit of 2200°F (Ref. 2).

For transients which may be DNB limited, Reactor Coolant System (RCS) flow and $F_{\Delta H}^N$ are of first order importance. The limits on $F_{\Delta H}^N$ ensure that the DNB design basis is met for normal operation, operational transients, and any transient conditions arising from faults of moderate frequency. The DNB design basis

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

precludes DNB and is met by limiting the minimum DNBR to the 95/95 limit, [1.3] using the [W-3 R-grid] CHF correlation.This value provides a high degree of assurance that the hottest fuel rod in the core does not experience DNB.

The allowable $F_{\Delta H}^{N}$ increases with decreasing power level. This functionality in $F_{\Delta H}^{N}$ is included in the analyses which provide the Reactor Core Safety Limits of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits are modeled implicitly assume 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 and the axial peaking factors are direct LOCA safety analyses inputs. The LOCA analyses verify the acceptability of the resulting PCT (Ref. 2).

The fuel is protected in part by Technical Specifications that preserve safety and accident analyses initial condition criteria on power peaking: LCO 3.2.3, Axial Flux Difference, LCO 3.2.4, Quadrant Power Tilt Ratio, LCO 3.1.7 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 - $F_Q(Z)$.

$F_{\Delta H}^{N}$ and $F_Q(Z)$ are 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 I events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

The Nuclear Enthalpy Rise Hot Channel Factor is an initial condition of Design Basis Accidents and Transient Analyses that presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 2 (Ref. 4)

(continued)

BASES

LCOs

$F_{\Delta H}^{N}$ shall be maintained within the limits specified by the LCO relationship. The value of CFDH, the $F_{\Delta H}^{N}$ limit at RATED THERMAL POWER, and PFDH, the Power Factor multiplier, are 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 unit safety analyses.

A Power Factor multiplier, PFDH, in the equation reflects 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 [2]% for every 1% of RTP reduction in THERMAL POWER.

APPLICABILITY

The $F_{\Delta H}^{N}$ limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits (DNBR and PCT). Applicability in other MODES is not required because there is insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power. In other words, the design bases events that are sensitive to $F_{\Delta H}^{N}$ in other MODES (MODE 2 through 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.1.1

With $F_{\Delta H}{}^{N}$ exceeding its limit, the unit is allowed 4 hours to restore $F_{\Delta H}{}^{N}$ to within its limits. This restoration could, for example, involve re-aligning any misaligned rods or reducing power enough to bring $F_{\Delta H}{}^{N}$ within its power dependent limit. When the $F_{\Delta H}{}^{N}$ limits are exceeded, the DNBR limit would likely not be violated in steady state operation, since events which 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 were to occur. Thus, the Completion Time of 4 hours represents a short time period during which the probability of a DNB limiting event occurring, is insignificant.

A.1.2.1

Four hours are allowed to reduce THERMAL POWER to < 50% of RATED THERMAL POWER (RTP) by boration or movement of the Control Rods. Due to the functional relationship between power level and $F_{\Delta H}{}^{N}$, the limit will increase with power level reduction. This is acceptable since the DNB margin increases with a reduction in power level. The Completion Time of 4 hours represents a short time period during which the probability of a DNB limiting event occurring is insignificant.

A.1.2.2

If the power level was reduced in accordance with Required Action A.1.2.1, then the Power Range Neutron Flux -- High trip setpoint must also be reduced. The reduction in the Power Range Neutron Flux -- High trip setpoints to ≤ 55% of RTP is intended to guarantee that the core will not be operated at a power level that may compromise the DNB limits assumed in the safety analyses. The Completion Time of 8 hours is a reasonable time, based upon operating experience, to reduce the setpoint.

(continued)

BASES

ACTIONS (continued)

A.2 and A.3

Once the power level has been reduced to < 50% of RTP an incore flux map must be obtained and the measured value of $F_{\Delta H}^N$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task. The Completion Time of 24 hours is allowed due to the increase in DNB margin which is obtained at lower powers and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operational experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F_{\Delta H}^N$.

Verification that $F_{\Delta H}^N$ is within its limit after an out-of-limit occurrence, assures that the cause of exceeding the limit has been corrected and subsequent operation will proceed within the LCO limit. This demonstrates that the $F_{\Delta H}^N$ limit is within the LCO limits prior to exceeding 50% of RTP and 75% of RTP, and within 24 hours after THERMAL POWER is ≥ 95% of RTP.

B.1

When Required Actions cannot be completed within their required Completion Time, the unit must be placed in MODE 2 where the LCO is not applicable. Placing the unit in MODE 2 precludes the core power distribution from exceeding the design limits. The Completion Time of 6 hours provided to be in MODE 2 is a reasonable time, based on operating experience, to reach MODE 2 from full power conditions without challenging the safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

NOTES

It is a requirement that the measured value of $F_{\Delta H}^N$ must be multiplied by [1.058] to account for measurement uncertainty before making comparisons to the $F_{\Delta H}^N$ limit (Ref. 5).

The requirements of SR 3.0.4 are exempted since the unit must be in MODE 1 to perform surveillances which demonstrate that the LCO is met.

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 31 EFPD surveillance frequency is acceptable for the power distribution changes which result from fuel burnup. The frequency of 31 EFPD is short enough so that the $F_{\Delta H}^N$ limit is not exceeded for any significant period of operation.

After each refueling it is required to measure $F_{\Delta H}^N$ prior to exceeding 75% of RTP. This requirement exists to ensure that $F_{\Delta H}$ limits are met at the beginning of each cycle.

BASES

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. Title 10 Code of Federal Regulations 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 1974.

3. ANSI N18.2-1973, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, American National Standards Institute, August 6, 1973.

4. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988 (NRC letter).

5. [WCAP-7308-L-P-A, Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988.]

B 3.2 REACTOR

B 3.2.3 Axial Flux Difference (CAOC Methodology)

BASES

BACKGROUND

The purpose of the limits on the values of AXIAL FLUX DIFFERENCE (AFD) is to limit the amount of axial power distribution skewing to the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are maintained within safety analysis assumptions. 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 axial power distribution, called Constant Axial Offset Control (CAOC), involves maintaining the AFD within a tolerance band around a burnup-dependent target to minimize the variation of axial peaking factor and axial xenon distribution during unit maneuvers.

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 (greater than or equal to 210 steps withdrawn) for steady-state operation at high power levels. Power level should be as near RATED THERMAL POWER as practical. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER (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.

The 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 $F_{\Delta H}^{N}$ and Quadrant Power Tilt Ratio LCOs limit the radial component of the peaking factors.

APPLICABLE SAFETY ANALYSES

AFD is a measure of axial power distribution skewing to the top or bottom half of the core. AFD is very sensitive to many core related parameters such as control bank positions, core power level, axial burnup distribution,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

axial xenon distribution and, to a lesser extent, coolant temperature and boron concentrations.The allowed range of AFD is used in the nuclear design process to confirm that operation within these limits produce core peaking factors and axial power distributions that meet safety analysis requirements.

The CAOC methodology (Ref. 1, 2, and 3) entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for the cycle which maximizes unit flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on AFD assure that the $F_Q(Z)$ peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on AFD also limit the range of power distributions which are initial conditions to Condition II, III, and IV events. This insures that clad integrity is maintained for these accidents. The most important Condition IV event is Loss of Coolant Accident. The most important Condition III event is the Loss of Flow Accident. The most important Condition II events are Uncontrolled Bank Withdrawal and Boration/Dilution Accidents. Condition II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

AXIAL FLUX DIFFERENCE is a process variable which is monitored and controlled by the operator to ensure that axial power distributions assumed in safety analyses are valid; hence it satisfies selection Criterion 2 of the NRC Interim Policy Statement (Ref. 4).

LCOs

The shape of the power profile in the axial or 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 manual operation of the Chemical Volume Control System to change boron concentration. Signals are available to the operator

(continued)

BASES

LCOs (continued)

from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 5). Separate signals are taken from the top and bottom detectors. AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience this flux difference is converted to percent to provide flux difference units, %-delta-flux or % ΔI.

The required target band varies with axial burnup distribution which in turn varies with core average accumulated burnup. The target band defined in the CORE OPERATING LIMITS REPORT (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.

With THERMAL POWER $\geq$ 90% of RTP, AFD must be kept within the target band. With AFD outside the target band with THERMAL POWER $\geq$ 90% of RTP, assumptions of accident analysis may be violated at the condition of maximum heat generation.

The NOTES with Parts b and c of the LCO define cumulative penalty deviation time. Although it is intended that the unit will be operated with the AFD within the target band about the target flux difference, 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 will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which 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% and < 90% of RTP, a 1-hour cumulative penalty deviation time limit, cumulative during the previous 24 hours, is provided for operation outside of the target band but within the Acceptable Operation Limits provided in the COLR. For THERMAL POWER levels > 15% and < 50% of rated thermal power, deviations of the AFD outside of the target

(continued)

BASES

LCOs (continued)

band are less significant. The accumulation of one half minute penalty deviation time per one minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 50% of RTP, AFD is not a significant parameter in assumptions of the safety analysis and requires no limits. However, the xenon distribution produced at low THERMAL POWER affects the power distribution as power is increased. Preventing unanalyzed xenon and power distributions is done by limiting accumulated penalty deviation time.

The frequency of monitoring AFD by unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time which allows the operator to accurately assess the status of penalty deviation time.

The LCO requires operation within the bounds analyzed during the core design process. If AFD cannot be maintained within the LCO limits, core power reduction is required.

Violating the LCO on AFD could produce unacceptable consequences should a Condition II, III, or IV event occur while AFD is outside the limits.

Figure B 3.2.3-1 shows a typical target band and typical AFD Acceptable Operation Limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% of RTP. Above 50% of RTP, the combination of THERMAL POWER and core peaking factors are of primary importance in safety analyses.

Between 15 and 50% of RTP, this LCO is applicable to assure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% of RTP and for lower MODES the stored energy in the fuel and energy being transferred to the coolant is low. The value of AFD in these conditions does not effect the limiting accident consequences.

(continued)

BASES (continued)

APPLICABILITY (continued)

For surveillance testing of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours. Some deviation in AFD is required for doing the NIS calibration with the incore detector system. However, this calibration is performed infrequently (quarterly).

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% of RTP.

ACTIONS

A.1

With AFD outside the target band with THERMAL POWER $\geq$ 90% of RTP, assumptions of accident analysis may be violated at the condition of maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore AFD within the target band because xenon distributions have very little time to change in this period.

A.2

Reducing THERMAL POWER to < 90% of RTP within the Completion Time of 15 minutes places the core in a condition which has been analyzed, provided that AFD is within the Acceptable Operation Limits provided in the COLR.

B.1

If Action A.1 or A.2 is not completed within the required Completion Time, the xenon distribution will become skewed. With this as an initial condition, the assumption that less than 1 hour outside the target band with THERMAL POWER < 90% but $\geq$ 50% of RTP will not significantly affect the xenon distribution is no longer valid. Reduction to < 50% of RTP within the Completion Time of 15 minutes and compliance with LCO requirements for subsequent increases in THERMAL POWER will assure that acceptable xenon conditions are restored.

B.2

Reduction in the Power Range Neutron Flux--High trip setpoints to $\leq$ 55% of RTP is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 8

(continued)

BASES

ACTIONS

B.2 (continued)

hours for the setpoint reduction is sufficient considering the likelihood of a transient in this time period and the reduced importance of AFD at the reduced power level.

C.1.1

With THERMAL POWER < 90% but ≥ 50% of RTP, 1 hour outside the target band is allowed if AFD is within the Acceptable Operation Limits provided in the COLR. With AFD within these limits, the power distribution will be acceptable as an initial condition for accident analysis for the existing xenon condition. The 1-hour cumulative penalty deviation time limits the extent of xenon redistribution which can occur. Without this limitation, unanalyzed xenon axial distributions could result with xenon buildup and decay. The reduction to < 50% of RTP puts the reactor at a THERMAL POWER level where AFD is not a significant accident analysis parameter.

Power cannot be returned to ≥ 50% of RTP until the cumulative penalty time is < 1 hour in the previous 24 hours. Each minute of penalty time is not removed until 24 hours after it was accrued.

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. (Note that any AFD within the Target Band is acceptable regardless of its relationship to the Acceptable Operation Limits). The reduction to < 50% of RTP puts the reactor at a THERMAL POWER level where AFD is not a significant accident analysis parameter. The Completion Time of 30 minutes allows for a prompt, yet orderly reduction in power.

C.1.2

Reduction in the Power Range Neutron Flux--High trip setpoints to ≤ 55% of RTP is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 8 hours for the setpoint reduction is sufficient considering the likelihood of a transient in this time period and the reduced importance of AFD at the reduced power level.

(continued)

BASES (continued)

ACTIONS (continued)

C.2

If Action C.1.1 is not completed within the required Completion Time, the xenon distribution will become skewed. If Xenon distribution becomes skewed, the assumption that a penalty deviation of 1 hour for each 2 hours of operation outside the target band is acceptable at < 50% of RTP is no longer valid. Reduction to < 15% of RTP within the Completion Time of 9 hours and compliance with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER will assure that acceptable xenon conditions are restored.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Provisions for monitoring the AFD on an automatic basis are derived from the unit process computer through the AFD monitor alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% of RTP. During operation at THERMAL POWER levels < 90% and > 15% of RTP, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour in the previous 24 hours.

The surveillance verifies that AFD as indicated by the Nuclear Instrumentation System excore channels is within the target band and consistent with the status of the AFD monitor alarm. The surveillance frequency of 7 days is adequate considering that AFD is controlled by the operator and will be frequently observed, particularly when power changes or control banks move. Any deviation from the target band which is not alarmed should be readily noticed.

SR 3.2.3.2

With the AFD monitor alarm inoperable, AFD is monitored to detect operation outside of the target band and to compute penalty deviation time. During operation at $\geq$ 90% of RTP, AFD is monitored at a surveillance frequency of 15 minutes to provide a high level of assurance that AFD is within its limits at high THERMAL POWER levels. At < 90% of RTP the frequency is reduced to 1 hour since the AFD may deviate from the target band for up to one hour before corrective action is required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

Monitored and logged values are assumed to exist for the preceding interval in order to conservatively compute the penalty deviation time. It is expected that AFD would be monitored and logged more frequently in periods of operation where power level or control bank positions are changing to allow corrective measures when AFD is more likely to move outside the target band.

SR 3.2.3.3

Updating the target flux difference at a frequency of 31 Effective Full Power Days (EFPD) provides for changing the target flux difference which may change by a few percent in that period due to burnup. Performing SR 3.2.3.4 will also update the target flux difference.

Alternatively, linear interpolation between the most recent measurement and an end of cycle value provides a reasonable update since the AFD changes due to burnup will tend toward zero flux difference. Where the end of cycle AFD from the cycle nuclear design is different from 0%, it may be a better value for the interpolation.

SR 3.0.4 is not applicable since the unit must be in MODE 1 to perform SR 3.2.3.3.

SR 3.2.3.4

Measurement of target flux difference is accomplished by taking a flux map with the core having equilibrium xenon conditions, preferably at high power levels with control banks nearly withdrawn. The flux map results yield the equilibrium xenon and axial power distribution from which the target value can be determined. Core burnup will cause the target flux difference to vary slowly.

A remeasurement frequency of 92 EFPD adjusts the target flux difference for each excore channel to that which the core would assume in steady-state conditions which is the basis for CAOC. Remeasurement on the surveillance interval also establishes AFD target flux difference values which will include changes in incore-excore calibration which may have occurred in the interim.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.3.4

Prior to initial fuel load and after each refueling, the initial target flux difference valves shall be based on design predictions.

SR 3.0.4 is not applicable since the unit must be in MODE 1 to perform the surveillance.

REFERENCES

1. WCAP-8403 (nonproprietary), Power Distribution Control and Load Following Procedures, Westinghouse Electric Corporation, September 1974.
2. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC.), Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package, January 31, 1980
3. C. Eicheldinger to D. B. Vassallo (Chief of Light Water Reactors Branch, NRC), Letter NS-CE-687, July 16, 1975.
4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987
5. Watts Bar FSAR, Section [7.7.1]

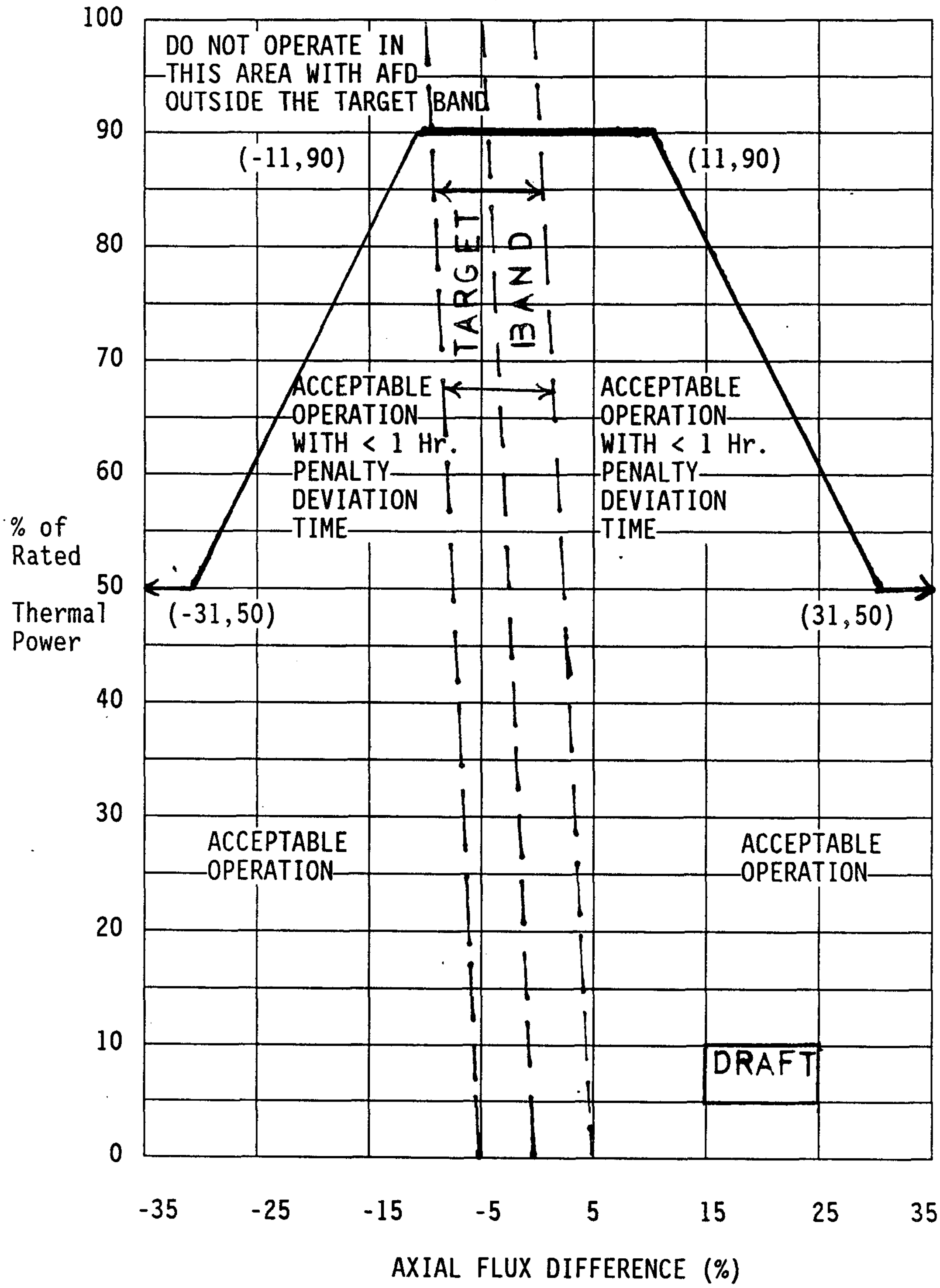


Figure B 3.2.3-1 (Page 1 of 1)

AXIAL FLUX DIFFERENCE Acceptable Operation Limits
and Target Band Limits as a Function
of RATED THERMAL POWER

B 3.2 REACTOR

B 3.2.4 Quadrant Power Tilt Ratio

BASES

BACKGROUND

The QUADRANT POWER TILT RATIO (QPTR) limit assures that the gross radial power distribution satisfies the design values used in the safety analysis. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

QUADRANT POWER TILT RATIO is a defined term in Section 1.1, Definitions.

The power density at any point in the core must be limited so that the fuel design criteria are maintained (Ref. 1 and 2). Together, LCO 3.2.3, Axial Flux Distribution (AFD), LCO 3.2.4, Quadrant Power Tilt Ratio (QPTR), and LCO 3.1.7, Control Bank Insertion Limits provide limits on process variables that characterize and control the 3-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 set by the safety analyses.

APPLICABLE SAFETY ANALYSES

The LCO limits on AFD, QPTR, $F_Q(Z)$, $F_{\Delta H}^N$ and control bank insertion are established to preclude core power distributions from occurring which would exceed the safety analyses limits:

QUADRANT POWER TILT RATIO limits assure 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.

QUADRANT POWER TILT RATIO is a process variable which is monitored and controlled by the operator to ensure that the core radial power distribution assumed in the accident analyses is valid; and as such it satisfies selection Criterion 2 of the NRC Interim Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCOs

The limit of 1.02, above which corrective action is required, provides DNBR and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F_{\Delta H}{}^N$ is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% of RATED THERMAL POWER (RTP) to preclude core power distributions from exceeding the design limits.

Applicability in MODE 1 $\leq$ 50% of RTP and in other MODES is not required because there is insufficient stored energy in the fuel or energy being transferred to the coolant to require the 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 LCOs still apply, but allow progressively higher peaking factors at 50% of RTP or below.

ACTIONS

A.1

With QPTR exceeding its limit, power level reduction of 3% from RTP for each 1% (i.e., each 0.01) that QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. For example, if QPTR = 1.03, the maximum allowable power level would be 91%. The Completion Time of 2 hours allows time to identify the cause and correct the tilt. Power reduction itself may cause a change in the tilted condition. Since the QPTR alarm would already be in its alarm state, additional changes in the QPTR would be detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR has increased, THERMAL POWER would have to be reduced accordingly.

A.2

Reduction in the Power Range Neutron Flux--High trip setpoints by at least 3% for each 1% that QPTR exceeds 1.0 is a conservative action for protection against the consequences of transients with unanalyzed power distributions. The Completion Time of 8 hours is allowed since the reduction in THERMAL POWER is the principal

(continued)

ACTIONS

A.2 (continued)

compensation for the tilted condition. Since the QPTR alarm would already be in its alarm state, additional changes in the QPTR would be detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR has increased, the setpoints would have to be reduced accordingly.

A.3

The peaking factors $F_{\Delta H}^N$ and $F_Q(Z)$ are of primary importance in assuring that the power distribution is within the initial conditions of the safety analysis. Performing Surveillance Requirements on $F_{\Delta H}^N$ and $F_Q(Z)$ within the Completion Time of 24 hours will assure that these primary indicators of power distribution are within their respective limits. If they are not, the Required Actions of these surveillances will provide additional core power reduction for the abnormal condition. If QPTR remains above its limits, 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. Changes would be expected due to burnup and xenon redistribution or correction of the cause of exceeding the QPTR limit.

A.4.1

Although $F_{\Delta H}^N$ and F_Q 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 safety analysis. A change in the power distribution can affect bank worths, peaking factors for rod malfunction accidents, etc. When 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 which merits investigation and evaluation. This begins with a examination of the incore power distribution, specifically core peaking factors and incore quadrant tilt because they characterize the actual state of the core power distribution. A reevaluation of each safety analysis in Table B 3.2.4-1 is performed prior to increasing THERMAL POWER to the RTP. This reevaluation is required to assure the core conditions for continued operation at RTP will be within safety analysis assumptions, and it must be completed prior to zeroing out the tilt.

(continued)

BASES

ACTIONS (continued)

A.4.2

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements will be met, the excore detectors should be recalibrated to show a zero QPTR prior to increasing THERMAL POWER. This is done to automatically detect any subsequent significant changes in QPTR.

B.1

When Required Actions cannot be completed within their required Completion Time, the only other acceptable action is to place the unit in a MODE where the LCO is not applicable. The LCO is not applicable with Thermal Power ≤ 50%. The Completion Time of 4 hours is based upon unit operating experience. This Completion Time will not place unnecessary stress on the unit systems or operators to reduce power to ≤ 50% from full power conditions.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

The surveillance required when the QPTR alarm is OPERABLE verifies that the QPTR as indicated by the NIS excore channels is within its limits. The surveillance frequency of 7 days is adequate given the low probability that the alarm would be inoperable without detection.

With the QPTR alarm inoperable, the Surveillance Frequency is increased to 12 hours. This frequency is adequate to detect any relatively slow changes in QPTR. For causes of tilt such as dropped rods which would occur quickly there will typically be other indications of abnormality which would prompt a verification of core tilt.

SR 3.2.4.2

With an NIS power range channel inoperable, tilt monitoring for a portion of the core is lost. Large tilts would likely be detected with the remaining channels, but the capability for tilt detection is degraded. Performing Surveillance Requirement 3.2.4.2 at a frequency of 12 hours provides an accurate alternative means for assuring that tilt remains within its limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

For purposes of monitoring QPTR when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and the previous data on the tilt. The incore detector monitoring is performed with a full incore flux map or two set of four symmetric thimbles. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for 3 and 4 Loop cores.

With one NIS channel inoperable the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has occurred which causes the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles or a complete flux map. Nominally, quadrant tilt from the surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. Title 10 Code of Federal Regulations 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors, 1974.

2. Title 10 Code of Federal Regulations 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987

Table 3.2.4-1 (Page 1 of 1)

Safety Analyses Requiring Reevaluation
in the Event of QPTR Exceeding 1.02

SAFETY ANALYSES
1. Rod Cluster Control Assembly Insertion Characteristics 2. Rod Cluster Control Assembly Misalignment 3. Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System 4. Single Rod Cluster Control Assembly Withdrawal at Full Power 5. Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident) 6. Major Secondary Coolant System Pipe Rupture 7. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System

BASES

BACKGROUND

The primary purpose of the Reactor Trip System (RTS) is to initiate a unit shutdown and/or to actuate the necessary safety system, based upon the values of selected unit parameters.

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below (refer to Figure B 3.3.1-1 for the discussion of the RTS):

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

- Signal process control and protection system, including analog protection system, Nuclear Instrumentation System, field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications.

- Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown/engineered safety features (ESF) actuation in accordance with the defined logic and based upon the bistable outputs from the signal process control and protection system.

- Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms and allow the rod cluster control assemblies (RCCAs) to fall into the core and shutdown the reactor. The bypass breakers allow at-power testing of the RTBs.

The OPERABILITY of the RTS is necessary to meet the assumptions of the safety analyses and provide for the mitigation, and in some cases, termination, of accident and transient conditions.

(continued)

This figure for illustration only. Do not use for operation.

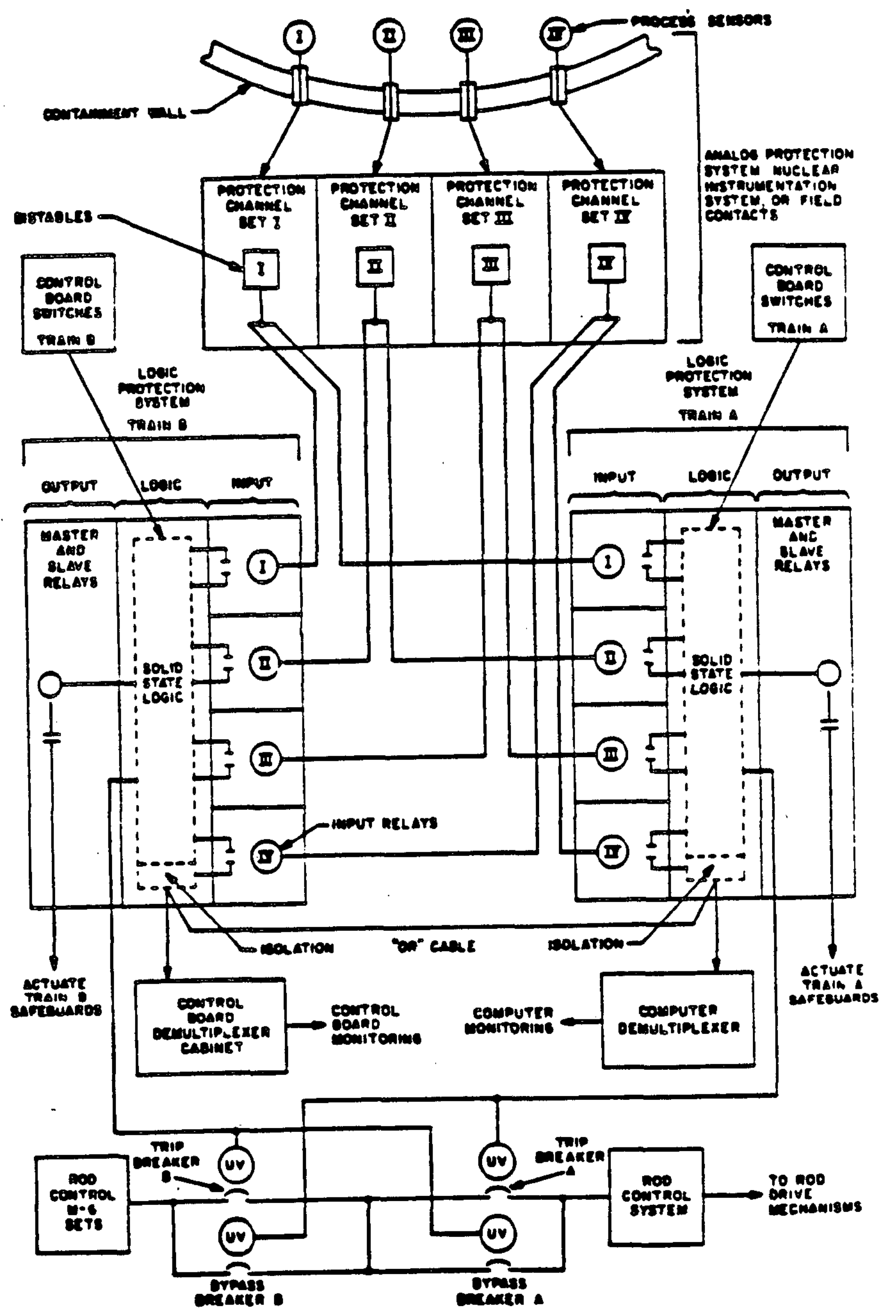


Figure B 3.3.1-1 (Page 1 of 1)
Reactor Trip System

BASES (continued)

BACKGROUND (continued)

Field Transmitters and Instrumentation

In order to meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the setpoint Allowable Value. The OPERABILITY of each transmitter can be evaluated when its "as found" value is compared against the Allowable Value.

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. 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 References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS while others provide input to the SSPS, main control board, unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a 2/3 logic are sufficient to provide the required reliability and redundancy. If one channel fails in the non-conservative direction, the function is still OPERABLE with a 2/2 logic. If one channel fails in the conservative direction, a trip will not occur because of the single failure and the function is still OPERABLE with a 1/2 logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a 2/4 logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system which may then require the protection function actuation, and a single failure in

(continued)

BASES (continued)

BACKGROUND (continued)

the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in Reference 4. The actual number of channels required for each unit parameter is specified in Reference 2.

The Trip Setpoints noted in Table 3.3.1-1 are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for channel calibration accuracy, i.e., ± (Rack Calibration + Comparator Setting Accuracy).

Note that LCO 3.3.1 includes the Limiting Safety System Settings (LSSS) as the Trip Setpoint values of Table 3.3.1-1. These values are established to prevent violation of the Safety Limits during normal unit operation and anitcipated operational occurrences.

Each channel of the process control equipment can be tested on line to verify that the signal/setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. Surveillance Requirements for the channels are specified in the Surveillance Requirements section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based upon the methodology described in Reference 5, 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.

Solid State Protection System (SSPS)

The SSPS equipment is used for the decision logic processing of bistable outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. In the event that one train is taken out of service for maintenance or test purposes, the second train will provide

(continued)

BASES (continued)

BACKGROUND (continued)

reactor trip/ESF actuation for the unit. In the event that both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements set down by the regulatory guides. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS provides the decision logic which warrants a reactor trip or ESF actuation, provides the electrical output signal which will initiate the required trip or actuation, and provides the status, interlock, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices which represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will initiate a reactor trip and/or send actuation signals via master and slave relays to those components whose aggregate function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the section on Applicable Safety Analyses.

Reactor Trip Switchgear

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 the power to the CRDMs which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. The output from the SSPS is a voltage signal which energizes the undervoltage coils in the RTBs, and bypass breakers if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are deenergized, the breaker trip lever is actuated by the deenergized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the deenergization of the undervoltage coils, each breaker is also equipped with a shunt trip device which is energized to trip the breaker open upon receipt of a reactor trip signal

(continued)

BASES (continued)

BACKGROUND (continued)

from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix functions are described in the FSAR Section [7.2]. In addition to the reactor trip/engineered safety features, these figures also describe the various "permissive interlocks" which are associated with unit conditions. Each train has a built-in testing device which can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

Design Basis Definition

The RTS and interlocks are designed to ensure that the following operational criteria are met:

- The associated actuation will occur when the parameter monitored by each channel reaches, or exceeds, its setpoint and the specified coincidence logic is satisfied;
- Separation and redundancy is maintained to permit a channel or train to be out of service for testing or maintenance while still maintaining reliability within the RTS instrumentation network;
- Functional capability is assured from diverse parameters.

Each of the analyzed accidents/transients can be detected by one or more RTS functions. The safety analyses take credit for most of the RTS functions. Those RTS functions for which no credit is taken in the safety analyses are still required to be OPERABLE because their functional capability at the specified Trip Setpoints enhances the overall diversity of the RTS.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

The RTS is part of the primary success path which functions to actuate or mitigate a MODE 1, 2, 3, or 4 DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement (Ref. 6).

REACTOR TRIPS

The reactor trip functions are designed to ensure that the reactor core is not operated beyond the design limits during steady-state operation, operational and anticipated transients, and postulated accidents. The reactor trip functions mitigate damage to the fuel due to overheating and subsequent cladding perforation which would release fission products to the reactor coolant.

1. Manual Reactor Trip

The control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A manual reactor trip accomplishes the same results as any one of the automatic trip functions. The safety analyses do not take credit for the Manual Reactor Trip. Manual Initiation of a reactor trip must be OPERABLE in MODES 1 and 2 which are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. The Manual Initiation function must also be OPERABLE in MODES 3, 4, and 5 if the shutdown rods or control rods are withdrawn or the Rod Control System is capable of withdrawing the shutdown rods and/or the control rods. 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. The core is also $\geq$ 5% shutdown as per LCO 3.9.1, Boron Concentration. Therefore, the manual initiation function does not have to be OPERABLE.

2.A Power Range, Neutron Flux--High Setpoint

The Power Range, Neutron Flux--High Setpoint trip function provides protection, from all power levels, against a positive reactivity excursion during power operations. The Nuclear Instrumentation System (NIS) power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

Steam Generator Water 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. 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 Power Range, Neutron Flux--High Setpoint trip must be OPERABLE in MODES 1 and 2 when a positive reactivity excursion could occur. This function will terminate the reactivity excursion and shutdown the reactor prior to reaching a power level that could damage the fuel. The Power Range, Neutron Flux-High Setpoint does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is shutdown and a reactivity excursion in the power range cannot occur. Other RTS functions and administrative controls provide protection against reactivity additions when in MODES 3, 4, 5, and 6. In addition, the NIS power range detectors cannot detect neutron levels in this range.

2.B Power Range, Neutron Flux--Low Setpoint

The Power Range, Neutron Flux--Low Setpoint trip function provides protection against a positive reactivity excursion from low power or subcritical conditions. The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

The Power Range, Neutron Flux--Low Setpoint trip must be OPERABLE in MODE 1 below the P-10 (Power Range Neutron Flux) setpoint and in MODE 2. This function may be manually blocked by the operator when 2/4 power range channels are greater than approximately 10% of RTP (P-10 setpoint). This function is automatically unblocked when 3/4 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 Setpoint trip function.

The Power Range, Neutron Flux--Low Setpoint trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is shutdown 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/power excursions in MODES 3, 4, 5, and 6.

2.C f(ΔI)

The f(ΔI) function is used in the calculation of the Overtemperature ΔT trip. It is a function of the indicated difference between the upper and lower NIS power range detectors. This function measures the axial power distribution. The ΔT 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.

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

The f(ΔI) function must be OPERABLE in MODES 1 and 2 when the Overtemperature ΔT trip is required to be OPERABLE as the f(ΔI) function provides one of the inputs to the Overtemperature ΔT trip.

3.A Power Range, Neutron Flux-High Positive Rate

The Power Range, Neutron Flux-High Positive Rate trip function provides protection against rapid increases in neutron flux which are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This function compliments the Power Range, Neutron Flux-High and Low Setpoint trip functions to ensure that the criteria are met for a rod ejection from the power range.

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Power Range, Neutron Flux-High Positive Rate trip must be OPERABLE in MODES 1 and 2 when there is a potential to add a large amount of positive reactivity from a rod ejection accident. The Power Range-Neutron Flux-High Positive Rate trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because other RTS trip functions and administrative controls will provide protection against positive reactivity additions. Also, only the shutdown rods may be withdrawn in MODES 3, 4, or 5 which limits the number of rods that are withdrawn and increases the net negative reactivity in the core. In MODE 6, no rods are withdrawn and the SHUTDOWN MARGIN is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels in this range.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

3.B Power Range, Neutron Flux-High Negative Rate

The Power Range, Neutron Flux-High Negative Rate trip function provides protection for multiple rod drop accidents. At high power levels a multiple rod drop accident could cause local flux peaking which would result in an unconservative local departure from nucleate boiling ratio (DNBR). DNBR is defined as the ratio of the heat flux required to cause a departure from nucleate boiling (DNB) at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this function for those rod drop accidents in which the local DNBRs will be greater than the limit.

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Power Range, Neutron Flux-High Negative Rate trip must be OPERABLE in MODES 1 and 2 when there is a potential for a multiple rod drop accident to occur. The Power Range, Neutron Flux-High Negative Rate trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the core is not critical and DNBR is not a concern. Also, only the shutdown rods may be withdrawn in MODES 3, 4, or 5 which limits the number of rods that are withdrawn and increases the net negative reactivity in the core. In MODE 6, no rods are withdrawn and the SHUTDOWN MARGIN is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels in this range.

4. Intermediate Range, Neutron Flux

The Intermediate Range, Neutron Flux trip function provides protection against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. This

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

trip function provides redundant protection to the Power Range, Neutron Flux--Low Setpoint trip function. The NIS Intermediate Range detectors 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. The safety analyses do not take credit for the Intermediate Range, Neutron Flux trip function. Even though the safety analyses take no credit for the Intermediate Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. Therefore, this function satisfies the requirements of IEEE 279 (Ref. 4) with a 1/2 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the control room operator when above the P-10 setpoint, which is 2/4 NIS power range channels greater than approximately 10% power, and is automatically unblocked when below the P-10 setpoint, which is 3/4 NIS power range channels less than approximately 10% power. 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 Intermediate Range, Neutron Flux trip must be OPERABLE in MODE 1 below the P-10 setpoint and in MODE 2 when there is a potential for an uncontrolled rod withdrawal accident during reactor startup. Above the P-10 setpoint, the Power Range, Neutron Flux--High Setpoint trip and the Power Range, Neutron Flux-High Positive Rate trip provide core protection for a rod withdrawal accident. The Intermediate Range, Neutron Flux trip does not have to be OPERABLE in MODES 3, 4, or 5 because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SHUTDOWN MARGIN to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has an increased SHUTDOWN MARGIN. Also, the NIS intermediate range detectors cannot detect neutron levels in this range.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

5. Source Range, Neutron Flux

The Source Range, Neutron Flux trip function provides protection against an uncontrolled bank rod withdrawal accident from a subcritical condition during startup. This trip function provides redundant protection to the Power Range, Neutron Flux--Low Setpoint, and Intermediate Range, Neutron Flux trip functions. Administrative controls also prevent the uncontrolled withdrawal of rods in MODES 3, 4, and 5. The NIS source range detectors 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 safety analyses do not take credit for the Source Range, Neutron Flux trip function. Even though the safety analyses take no credit for the Source Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint is assumed to be available and the trip is implicitly assumed in the safety analyses. Therefore, this function satisfies the requirements of IEEE 279 (Ref. 4) with a 1/2 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. This trip can be manually blocked by the control room operator when above the P-6 setpoint (Intermediate Range Neutron Flux interlock) and is automatically unblocked when below the P-6 setpoint. The manual block of the trip function also deenergizes the source range detectors. The source range detectors are automatically re-energized when below the P-6 setpoint. The P-10 signal also provides a backup signal to deenergize the source range detectors.

The Source Range, Neutron Flux trip must be OPERABLE in MODE 2 when below the P-6 setpoint during a reactor startup. Above the P-6 setpoint, the Intermediate Range, Neutron Flux trip and the Power Range, Neutron Flux--Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are deenergized and inoperable. The Source Range, Neutron Flux

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

trip function must also be OPERABLE in MODES 3, 4, and 5 with the reactor shut down. If the Rod Control System is capable of rod withdrawal, the Source Range, Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the Rod Control System is not capable of rod withdrawal, the source range detectors 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.

6. Overtemperature ΔT

The Overtemperature ΔT trip function provides protection to prevent a departure from nucleate boiling (DNB). 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 all combinations of pressure, power, coolant temperature, and axial power distribution, assuming full reactor coolant flow. Protection from DNB is assured provided that the transient is slow with respect to delays from the core to the measurement system, and pressure is between the Pressurizer Pressure--High and Low Trip Setpoints. The Overtemperature ΔT trip function uses each loop's ΔT as a measure of reactor power and is automatically varied with the following parameters:

- o 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,
- o Pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure,
- o Axial power distribution (discussed under Function 2.C, $f(\Delta I)$), and
- o Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature ΔT trip function is calculated for each loop as described in Note 1 of Table 3.3.1-1. The pressure and temperature signals are used for other control functions. Therefore, the actuation logic must be able to withstand an input failure to the control system which may

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This function satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment. 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 a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The Overtemperature ΔT trip must be OPERABLE in MODES 1 and 2 to prevent DNB. This trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip function provides protection to ensure the integrity of the fuel, i.e., no fuel pellet melting and less than 1% cladding strain, 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 and is automatically varied with the following parameters:

- o 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, and

- o The rate of change of reactor coolant average temperature and the delays between the core and the temperature measurement system are dynamically compensated for.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

The Overpower ΔT trip function is calculated for each loop as per Note 3 of Table 3.3.1-1. The temperature signals are used for other control functions. 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 remaining channels providing the protection function actuation. This function satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide protection for a steam line break accident and may be in a harsh environment. 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 a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The Overpower ΔT trip function must be OPERABLE in MODES 1 and 2. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. This trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure--Low

The Pressurizer Pressure--Low trip function provides protection against DNB due to low pressure. The pressurizer pressure channels are also used to provide input to the Pressurizer Pressure 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Pressurizer Pressure--Low trip must be OPERABLE in MODE 1 when DNB is a major concern. This trip function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip function is automatically blocked below P-7. Below the P-7 setpoint, no conceivable power distributions can occur that would cause DNB concerns.

9. Pressurizer Pressure--High

The Pressurizer Pressure--High trip function provides protection against overpressurizing the Reactor Coolant System (RCS). This trip function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions. The pressurizer pressure channels are also used to provide input to the Pressurizer Pressure 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 satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Pressurizer Pressure--High trip must be OPERABLE in MODES 1 and 2 to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. The Pressurizer Pressure--High trip function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because transients which could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

10. 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. The pressurizer level channels are used as input to the Pressurizer Level Control System. However, the level channels do not actuate the safety valves. Therefore, this function satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 logic. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Pressurizer Water Level--High trip must be OPERABLE in MODE 1 when there is a potential for overfilling the pressurizer. This trip function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip function is automatically blocked below P-7. Below the P-7 setpoint, transients which could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

11.A Reactor Coolant Flow--Low (Single Loop)

The Reactor Coolant Flow--Low (Single Loop) trip function provides protection against DNB due to low flow in one or more RCS loops. Above the P-8 setpoint which is approximately 48% of RTP, a loss of flow in any 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. Therefore, this function satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 logic in any RCS loop. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Reactor Coolant Flow--Low (Single Loop) trip must be OPERABLE in MODE 1 above the P-8 setpoint when a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint a loss of flow in two or more loops is required to

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

actuate a reactor trip because of the lower power level and the greater margin to DNB.

11.B Reactor Coolant Flow--Low (Two Loops)

The Reactor Coolant Flow--Low (Two Loops) trip function provides protection against DNB due to low flow in two or more RCS loops. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more 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. Therefore, this function satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 logic in 2/4 RCS loops. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Reactor Coolant Flow--Low (Two Loops) trip must be OPERABLE in MODE 1 above the P-7 setpoint and below the P-8 setpoint. Below the P-7 setpoint, 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 two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to DNB.

12. Undervoltage-Reactor Coolant Pumps

The Undervoltage-Reactor Coolant Pumps reactor trip function provides protection against DNB due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two RCP buses will actuate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low Trip Setpoint is reached. Time delays are incorporated into the Undervoltage-Reactor Coolant Pumps channels to prevent reactor trips due to momentary electrical power transients. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

The Undervoltage-Reactor Coolant Pumps trip must be OPERABLE in MODE 1 above the P-7 setpoint. 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 loss of flow in two RCS loops is automatically enabled.

13. Underfrequency-Reactor Coolant Pumps

The Underfrequency-Reactor Coolant Pumps reactor trip function provides protection against DNB due to a loss of flow in two or more RCS loops. The frequency to each RCP is monitored. Above the P-7 setpoint, a loss of flow in two or more loops will initiate a reactor trip. A decrease of the frequency detected on two RCP buses will actuate a reactor trip. This trip function will generate a reactor trip before the Reactor Coolant Flow--Low Trip Setpoint is reached. Time delays are incorporated into the Underfrequency-Reactor Coolant Pumps channels to prevent reactor trips due to momentary electrical power transients. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Underfrequency-Reactor Coolant Pumps trip must be OPERABLE in MODE 1 above the P-7 setpoint. 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 loss of flow in two RCS loops is automatically enabled.

14. Steam Generator Level--Low-Low

The Steam Generator Water Level--Low-Low trip function provides protection against a loss of heat sink and ensures a sufficient volume of water to allow starting the auxiliary feedwater system prior to uncovering the steam generator tubes. The steam generators are the heat sink for the reactor. In order to act as a heat sink, the steam generators must contain a minimum amount of water. The

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

purpose of this reactor trip is to provide reactor protection in the event of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a Steam Generator Level--Low Coincident With Steam/Feedwater Flow Mismatch reactor trip. A narrow range low-low level in any steam generator is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the Steam Generator Water Level Control System and the protection system. Normally, IEEE 279 (Ref. 4) would require four channels of instrumentation. However, the reactor trip on Steam Generator Level--Low Coincident With Steam/Feedwater Flow Mismatch also accomplishes the same function. A slow drift in the level signal may not actuate a Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch reactor trip. Since the level decrease is slow, the operator has time to respond to the low level alarms which occur prior to reaching the reactor trip. Since only one steam generator is affected, automatic protection is not mandatory and a reactor trip on 2/3 low-low level signals in any steam generator is acceptable. The Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties because this function provides primary protection for an event that results in a harsh environment. This function also performs the ESFAS function of starting the auxiliary feedwater pumps on low-low steam generator level.

The Steam Generator Water Level--Low-Low trip must be OPERABLE in MODES 1 and 2 when the reactor requires a heat sink. The normal source of water for the steam generators is the Main Feedwater System (non-safety related). The Main Feedwater System is normally in operation in MODES 1, 2, and 3. The Auxiliary Feedwater System is the safety related backup source of water to ensure that the steam generators remain the heat sink for the reactor. During normal startups and shutdowns, the Main Feedwater System provides feedwater to maintain steam generator level. The Steam Generator Water Level--Low-Low function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is not operating or even critical. Decay heat removal is accomplished by the Main Feedwater System in MODE 3 and the Residual Heat Removal System in MODES 4, 5, and 6.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

15. Steam Generator Water Level--Low, Coincident With Steam/Feedwater Flow Mismatch

This reactor trip function, in conjunction with the Steam Generator Water Level--Low-Low reactor trip, provides protection against a loss of heat sink and ensures a sufficient volume of water to allow startup of the Auxiliary Feedwater System prior to uncovering the steam generator tubes. In addition to a decreasing water level in the steam generator, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow. With less feedwater flow than steam flow, steam generator level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There is one steam generator level channel and two steam/feedwater flow mismatch channels per steam generator. One narrow range level channel sensing a low level Coincident With one steam/feedwater flow mismatch channel sensing a flow mismatch (steam flow greater than feed flow) will actuate a reactor trip. The Steam Generator Level--Low Trip Setpoint reflects only steady state instrument uncertainties because this function does not provide primary protection for any events that result in a harsh environment. The Steam/Feedwater Flow Mismatch Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any events that result in a harsh environment.

The Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch trip must be OPERABLE in MODES 1 and 2 when the reactor requires a heat sink. The normal source of water for the steam generators is the Main Feedwater System (non-safety related). The Main Feedwater System is in operation in MODES 1, 2, and 3. The Auxiliary Feedwater System is the safety related backup source of water to ensure that the steam generators remain the heat sink for the reactor. During normal startups and shutdowns, the Main Feedwater System provides feedwater to maintain steam generator level. The Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch function does not have to be OPERABLE in MODES 3, 4, 5, or 6 the reactor is not operating or even critical. Decay heat removal is accomplished by the Residual Heat Removal System in MODES 4, 5, and 6.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

16.A Turbine Trip-Low Fluid Oil Pressure

The Turbine Trip-Low Fluid Oil Pressure trip function is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. This trip function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. The safety analyses do not take credit for this reactor trip function. Three pressure switches monitor the control oil pressure in the turbine Electrohydraulic Control System. A low pressure sensed by 2/3 of the pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit 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 Turbine Trip-Low Fluid Oil Pressure trip must be OPERABLE in MODE 1 above the P-9 setpoint. Below the P-9 setpoint, a turbine trip can be accomodated by the Steam Dump System. There is no potential for a turbine trip in MODES 2, 3, 4, 5, or 6 and the Turbine Trip-Low Fluid Oil Pressure trip function does not need to be OPERABLE.

16.B Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip function is anticipatory for the loss of heat removal capabilities of the secondary system following a turbine trip. This trip functions to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. The safety analyses do not take credit for this function. The unit 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

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

the Pressurizer Pressure--High trip function and RCS integrity is ensured by the pressurizer safety valves. This trip function is redundant to the Turbine Trip-Low Fluid Oil Pressure trip function. Each turbine stop valve is equipped with a limit switch. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

The Turbine Trip-Turbine Stop Valve Closure trip must be OPERABLE in Mode 1 above the P-9 setpoint. Below the P-9 setpoint, a load rejection can be accomodated by the Steam Dump System. There is no potential for a load rejection in MODES 2, 3, 4, 5, or 6 and the Turbine Trip-Turbine Stop Valve Closure trip function does not need to be OPERABLE.

17. Safety Injection Input From ESF

If a reactor trip has not already been generated by the Reactor Trip System, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal which initiates Safety Injection. The ESF systems, when utilized to the analyzed capabilities, will ensure that the reactor is shutdown. This is a condition of acceptability for the 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 which is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time a Safety Injection signal is present. This function must be OPERABLE in MODES 1 and 2 when the reactor is critical and must be shutdown in the event of an accident. In MODES 3, 4, 5, and 6, the reactor is not critical and this function does not need to be OPERABLE.

REACTOR TRIP SYSTEM INTERLOCKS

18.A Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6, interlock is actuated when either NIS intermediate range channel goes approximately one decade above the minimum channel reading. The P-6 interlock performs the following functions:

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

- On increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. 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.

- On decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range, Neutron Flux reactor trip.

The P-6 interlock must be OPERABLE in MODE 2 when below the P-6 interlock setpoint. Above the P-6 interlock setpoint, the NIS Source Range, Neutron Flux reactor trip will be blocked and this function will no longer be necessary. The P-6 interlock does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the NIS Source Range is providing core protection.

18.B Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7, interlock is actuated by input from either the P-10, Power Range Neutron Flux, OR the P-13, Turbine Impulse Chamber Pressure, interlocks. The P-7 interlock provides the following functions:

- On increasing power, the P-7 interlock automatically enables reactor trips on the following functions:

  Pressurizer Pressure--Low,
  Pressurizer Water Level--High,
  Reactor Coolant Flow--Low (Two Loops),
  Undervoltage-Reactor Coolant Pumps, and
  Underfrequency-Reactor Coolant Pumps,

These reactor trips are only required when operating above about 10% power, the P-7 setpoint. These reactor trips provide protection against DNB. Below 10% power, the RCS is capable of providing sufficient natural circulation without any reactor coolant pumps running to prevent DNBR.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

- On decreasing power, the P-7 interlock automatically blocks reactor trips on the following functions:

  Pressurizer Pressure--Low,
  Pressurizer Water Level--High,
  Reactor Coolant Flow--Low (Two Loops),
  Undervoltage-Reactor Coolant Pumps, and
  Underfrequency-Reactor Coolant Pumps,

The Low Power Reactor Trips Block, P-7, interlock must be OPERABLE in MODE 1. The low power trips are blocked below 10% power and unblocked above 10% power. This function does not have to be OPERABLE in MODES 2, 3, 4, 5, or 6 because the interlock performs its function when power level drops below 10% power which is in MODE 1.

18.C Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8, interlock is actuated at approximately 48% power as determined by 2/4 NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow-Low (Single Loop) reactor trip on low flow in one or more RCS loops on increasing power. A loss of flow in one RCS loop could result in DNB conditions in in the core when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in one loop is automatically blocked.

The Power Range Neutron Flux, P-8, interlock must be OPERABLE in MODE 1 when a loss of flow in one RCS loop could result in DNB conditions. This function does not have to be OPERABLE in MODES 2, 3, 4, 5, or 6 because the core is not producing sufficient power to be concerned about DNB conditions.

18.D Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9, interlock is actuated at approximately 50% power as determined by 2/4 NIS power range detectors. The P-9 interlock automatically enables the Turbine Trip-Low Fluid Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when above the P-9 setpoint to minimize the transient on the reactor. The Power Range Neutron Flux, P-9, Interlock must be OPERABLE in MODE 1 when a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System. This function does not have to be OPERABLE in MODES 2, 3, 4, 5, or 6 because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

18.E Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10, interlock is actuated at approximately 10% power as determined by 2/4 NIS power range detectors. The P-10 interlock performs the following functions:

- On increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range, Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal.
- On increasing power, the P-10 interlock allows the operator to manually block the Power Range, Neutron Flux--Low Setpoint reactor trip.
- On increasing power, the P-10 interlock automatically provides a backup block signal to the Source Range, Neutron Flux reactor trip and also to deenergize the NIS source range detectors.
- On increasing power, the P-10 interlock provides one of the two inputs to the P-7 interlock.
- On decreasing power, the P-10 interlock automatically enables the Power Range, Neutron Flux--Low Setpoint reactor trip and the Intermediate Range, Neutron Flux reactor trip and rod stop.

The Power Range Neutron Flux, P-10, interlock must be OPERABLE in MODE 1 when the reactor is at power. This function must also be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

Power Range, Neutron Flux--Low Setpoint and Intermediate Range, Neutron Flux reactor trips. This function does not have to be OPERABLE in MODES 3, 4, 5, or 6 because the reactor is not at power and the Source Range, Neutron Flux reactor trip provides core protection.

18.F Turbine Impulse Chamber Pressure, P-13

The Turbine Impulse Chamber Pressure, P-13, interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by 1/2 pressure detectors. The P-13 interlock only provides input to the P-7 interlock.

The P-13 interlock must be OPERABLE in MODE 1 when the turbine generator is operating. This function does not have to be OPERABLE in MODES 2, 3, 4, 5, or 6 because the turbine generator is not operating.

19. Reactor Trip Breakers, and
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The RTBs provide the means to interrupt the power to the control rod drive mechanisms and 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 equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS cause the RTBs and associated bypass breakers to open and shutdown the reactor. the RTBs and bypass breakers if in use, must be OPERABLE in MODES 1 and 2 when the reactor is critical. The RTBs, and bypass breakers if in use, must be OPERABLE in MODES 3, 4, and 5 if they are closed and the Rod Control System is kept subcritical by the increased SHUTDOWN MARGIN.

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BASES (continued)

ACTIONS

In the event a channel's Trip Setpoint is found non-conservative with respect to the Allowable Value; or the transmitter, a rack module, or an SSPS module is found inoperable, then the function which that channel provides must be declared inoperable and the unit must enter the Condition statement for the particular protection function affected. If the number of inoperable channels or trains for a particular protection function is greater than the number of inoperable channels addressed by the Condition statement, then the unit no longer meets the assumptions of the safety analyses. The unit must then be placed in a MODE where the function is no longer required to be OPERABLE as per LCO 3.0.3.

Condition A

Condition A is applicable to all RTS protection functions. The most frequent occurrence to render a protection function inoperable is the determination that a bistable or process module has drifted sufficiently to exceed the Allowable Value. Typically, the drift is not large, which would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the Allowable Value, the channel must be declared inoperable immediately and the appropriate Condition from Table 3.3.1-1 must be entered. Condition A addresses the situation where one or more channels for one or more functions are inoperable at the same time. 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.

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BASES (continued)

ACTIONS (continued)

Condition B

Condition B applies to the Manual Reactor Trip in MODES 1 and 2. This action addresses the train orientation of the SSPS for this function. If a train is inoperable, 48 hours is allowed to return it to an OPERABLE status. If the function can not be returned to an OPERABLE status, six additional hours are allowed to place the unit in MODE 3, and one additional hour is allowed to open the RTBs. The Completion Time of six hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. With the RTBs open and the unit in MODE 3, this trip function is no longer required to be OPERABLE. The allowance of 48 hours to return the train to an OPERABLE status and six hours to reach MODE 3 are justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. The additional hour to open the RTBs is based on engineering judgement. Allowance of these intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition C

Condition C applies to the following reactor trip functions in MODES 3, 4, and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal:

- Manual Reactor Trip,
- Reactor Trip Breakers,
- Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, and
- Automatic Trip Logic

This action addresses the train orientation of the SSPS for these functions. If a train is inoperable, 48 hours is allowed to return it to an OPERABLE status. If the function can not be returned to an OPERABLE status, one additional hour is allowed to have the RTBs open. With the RTBs open, these functions are no longer required. The allowance of 48 hours to return the train to an OPERABLE status and one hour to have the RTBs open are justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

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BASES (continued)

ACTIONS (continued)

Condition D

Condition D applies to the following reactor trip functions:

- Power Range, Neutron Flux--High Setpoint,
- Power Range, Neutron Flux-High Positive Rate, and
- Power Range, Neutron Flux-High Negative Rate,

Six hours is allowed to place the channel in a tripped condition. The NIS power range detectors provide input to the Rod Control System and the Steam Generator Water Level Control System and therefore have a 2/4 trip logic. A known inoperable channel must be placed in the tripped condition in order to allow for two subsequent failures in the three remaining OPERABLE channels. This results in a partial trip condition requiring only 1/3 logic for actuation. The six hours allowed to place the inoperable channel in the tripped condition and the four hours allowed for this channel to be in the bypassed condition for testing and for resetting the Trip Setpoint of other channels is justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% of RTP and the Power Range, Neutron Flux--High Setpoint must be reduced to ≤ [85]% of RTP within four hours. While the Trip Setpoints of the OPERABLE channels are being reduced, the inoperable channel should be placed in the bypass condition to prevent an inadvertent reactor trip when the OPERABLE channels are placed in the trip condition to reduce their Trip Setpoints. Reducing the power level and resetting the Trip Setpoint prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, one-fourth of the radial power distribution monitoring capability, Channel Deviation, is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition and the QUADRANT POWER TILT RATIO (QPTR) monitored every 12 hours as per SR 3.2.4.2. Calculating the QPTR every 12 hours compensates for the lost monitoring capability due to the

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ACTIONS (continued)

inoperable NIS power range detector and allows continued unit operation at power levels > 75% of RTP. The four hour and 12 hour frequencies are consistent with LCO 3.2.4, Quadrant Power Tilt Ratio.

Condition E

Condition E applies to the following reactor trip functions:

- Power Range, Neutron Flux--Low Setpoint,
- f(ΔI),
- Overtemperature ΔT,
- Overpower ΔT,
- Pressurizer Pressure--Low,
- Pressurizer Pressure--High,
- Reactor Coolant Flow--Low
- Undervoltage-Reactor Coolant Pumps,
- Underfrequency-Reactor Coolant Pumps,
- Pressurizer Water Level--High,
- Steam Generator Water Level--Low Low, and
- Turbine Trip-Low Fluid Oil Pressure

The Pressurizer Water Level--High, Steam Generator Water Level--Low Low, Reactor Coolant Flow--Low, and Turbine Trip-Low Fluid Oil Pressure trips are 2/3 coincidence logic. All the rest are 2/4 coincidence logic. A known inoperable channel must be placed in the tripped condition within 6 hours. This results in a partial trip condition requiring only 1/2 logic for actuation of the 2/3 trips and 1/3 logic for actuation of the 2/4 trips. The six hours allowed to place the inoperable channel in the tripped condition and the four hours allowed for this channel to be in the bypassed condition for testing are justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition F

Condition F applies to the Intermediate Range, Neutron Flux trip when below the P-6 setpoint. Even though the safety analyses take no credit for the Intermediate Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. Below the P-6 setpoint, the NIS source range performs the

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BASES (continued)

ACTIONS (continued)

monitoring and protective functions. If THERMAL POWER is < the P-6 setpoint, the inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. The Completion Time is based on the requirement to have the NIS intermediate range OPERABLE in this unit condition.

Condition G

Condition G applies to the Intermediate Range, Neutron Flux trip when above the P-6 setpoint and below the P-10 setpoint. Even though the safety analyses take no credit for the Intermediate Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range performs the monitoring functions. If THERMAL POWER is > the P-6 setpoint but ≤ the P-10 setpoint, two hours is allowed to return the inoperable channels to OPERABLE status <u>OR</u> THERMAL POWER must be reduced to < the P-6 setpoint <u>OR</u> increased to ≥ the P-10 setpoint within two hours. The NIS Intermediate Range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is ≥ the P-10 setpoint, the NIS power range detectors perform the monitoring and protective functions and the intermediate range is not required. The Completion Times are based on industry operating experience and engineering judgement.

Condition H

Condition H applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 above the P-6 setpoint and below the P-10 setpoint. Even though the safety analyses take no credit for the Intermediate Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range performs the monitoring functions. With no intermediate range channels OPERABLE, the Required

(continued)

BASES (continued)

ACTIONS (continued)

Actions are to suspend operations involving positive reactivity additions within 15 minutes. This will preclude any power level increase since there are no OPERABLE neutron flux channels. The Completion Time of 15 minutes is the shortest time allowed for manual operator actions. The operator must also reduce power to below the P-6 setpoint within two hours so that the Source Range channels will be able to monitor the core power level. The Completion Time of two hours is based on engineering judgement and will allow a slow and controlled power reduction to less than the P-6 setpoint.

Condition I

Condition I applies to one inoperable Source Range, Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. Even though the safety analyses takes no credit for the Source Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of RTS. With the unit in this condition, < P-6, the NIS source range performs the monitoring and protective functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended within 15 minutes. This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended as quickly as possible. The Completion Time of 15 minutes is based on engineering judgement, industry operating experience, and the fact that the short time duration would not add significantly to the probability of an accident.

Condition J

Condition J applies to two inoperable Source Range, Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. Even though the safety analyses takes no credit for the Source Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of RTS. With the unit in this condition, < P-6, the NIS source range performs the

(continued)

BASES (continued)

ACTIONS (continued)

monitoring and protective functions. With both source range channels inoperable, operations involving positive reactivity additions shall be suspended within 15 minutes. With no source range channels OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended as quickly as possible. In addition to suspending positive reactivity additions, the RTBs must be opened within 15 minutes. With the RTBs open and positive reactivity additions suspended, the core is in a relatively safe and stable condition. With the RTBs open, the unit enters Condition L and the clock starts at time zero. The Completion Times of 15 minutes to suspend positive reactivity additions and open the RTBs is based on engineering judgment and industry operating experience.

Condition K

Condition K applies to one inoperable source range channel in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. Even though the safety analyses takes no credit for the Source Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of RTS. With the unit in this condition, < P-6, the NIS source range performs the monitoring and protective functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel can not be returned to an OPERABLE status, one additional hour is allowed to have the RTBs open. In addition to opening the RTBs, all operations involving positive reactivity additions must be suspended within the same hour. Suspension of positive reactivity additions and opening the RTBs will preclude any power excursion. Also, all valves that could add unborated water to the RCS must be closed within the same one hour as per Required Action A.2 of LCO 3.9.2, Unborated Water Source Isolation Valves. The isolation of unborated water sources will preclude a boron dilution accident. The allowance of 48 hours to restore the channel to OPERABLE status and the additional hour to open the RTBs are justified in Reference 7. The allowance of one hour to stop positive reactivity additions and close the unborated water source isolation valves is based on engineering judgement and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition L

Condition L applies to no OPERABLE Source Range, Neutron Flux trip channels when in MODES 3, 4, or 5 with the RTBs open. Even though the safety analyses take no credit for the Source Range, Neutron Flux trip, the functional capability at the specified Trip Setpoint enhances the overall diversity of the RTS. With the unit in this condition, the NIS source range performs the monitoring and protective functions. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended within 15 minutes. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within one hour as per Required Action A.2 of LCO 3.9.2, Unborated Water Source Isolation Valves. The isolation of unborated water sources will preclude a boron dilution accident. Also, the SHUTDOWN MARGIN (SDM) must be verified within one hour and once per 12 hours as per SR 3.1.1.1. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within one hour allows sufficent time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions, therefore, core reactivity should not be increasing and a 12 hour frequency is adequate. The Completion Times of within one hour and once per 12 hours are based on engineering judgment and the fact that unit conditions will be changing very slowly.

Condition M

Condition M applies to the Steam Generator level--Low Coincident with Steam/Feedwater Flow Mismatch reactor trip. A known inoperable channel must be placed in the tripped condition within six hours. This results in a partial trip condition. The six hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

(continued)

BASES (continued)

ACTIONS (continued0

Condition N

Condition N applies to the Safety Injection Input From ESF reactor trip. This action addresses the train orientation of the RTS for this function. With one train inoperable, there is no action that may be taken (other than restoration to an OPERABLE status) to allow an extended period of unit operation without a significant increase in the probability of a core melt. Therefore, the Required Action is to commence an orderly shutdown with the unit in MODE 3 within six hours. The Completion Time of six hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Placing the unit in MODE 3 removes the requirement for this particular function. The two hours that the inoperable train may be in the bypassed condition for testing is justified in Reference 7, evaluated as Test Time in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition P

Condition P applies to the P-6, P-7, P-8, P-9, P-10 (above the P-10 setpoint), and P-13 interlocks. With one channel inoperable, the associated interlock should be verified to be in its required state for the existing plant condition within one hour. The Completion Time of one hour is based on engineering judgment and the minimum amount of time allowed for manual operator actions.

Condition Q

Condition Q applies to the P-10 input to the P-7 interlock, and the P-10 interlock when below the P-10 setpoint. With two channels inoperable for 2/4 coincidence logic, the associated interlock should be verified to be in its required state for the existing plant condition within one hour. These actions are conservative for the case where power level is being raised. The Completion Time of one hour is based on engineering judgment and the minimum amount of time allowed for manual operator actions.

(continued)

BASES (continued)

ACTIONS (continued)

Condition R

Condition R applies to the Reactor Trip Breakers and the RTS Automatic Trip Logic in MODES 1 and 2. This action addresses the train orientation of the RTS for these functions. With one train inoperable, there is no action that may be taken (other than restoration to an OPERABLE status) to allow an extended period of unit operation without a significant increase in the probability of a core melt. Therefore, the Required Action is to commence an orderly shutdown with the unit in MODE 3 within six hours.

The Completion Time of six hours to reach MODE 3 from MODE 1 is reasonable based on operating experience and normal cooldown rates, and does not challenge safety systems or operators. Placing the unit in MODE 3 transfers applicability to Condition C if the RTBs are closed and the Rod Control System is capable of rod withdrawal. The two hours allowed for these functions to be in the bypassed condition for testing is justified in Reference 7, evaluated as Test Time in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition S

Condition S applies to the Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the breaker(s) must be declared inoperable. With the breaker inoperable, Condition R applies. Note that entry into Condition R from this point starts the clock at time zero. The affected breaker shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to restore the breaker to an OPERABLE status. The allowance of 48 hours to restore the breaker to an OPERABLE status is justified in GL 85-09 (Ref. 8) and does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition T

Condition T applies to the Turbine Trip-Turbine Stop Valve Closure reactor trip function, in MODE 1. With one channel inoperable, it must be restored to OPERABLE status within 48 hours or reduce THERMAL POWER to < P-9 setpoint within the following 6 hours.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The Surveillance Requirements for any particular RTS function are found in the Surveillance Requirements column of Table 3.3.1-1, for that function.

Note that <u>each</u> channel of process protection supplies <u>both</u> trains of the RTS. When testing Channel I, Train A <u>and</u> Train B must be examined. Similarly, Train A <u>and</u> Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TESTS are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions, e.g., lag, lead/lag, rate/lag, etc., the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate technical Specification response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

<u>SR 3.3.1.1</u>

Surveillance Requirement 3.3.1.1 is the performance of a CHANNEL CHECK. A CHANNEL CHECK is simply the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the three or four channels indicated in the control room should be reading approximately the same. Agreement is determined by the unit staff based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the racks have drifted outside of their limit. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their Trip Setpoints. The surveillance interval, about once every shift, is based on engineering judgment.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

Surveillance Requirement 3.3.1.2 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 24 hours when greater than 15% of RTP. The outputs of the NIS Power Range channels are normalized to the results of the calorimetric. The CHANNEL CALIBRATION consists only of a comparison of the results of the calorimetric with the NIS channel output. This surveillance is not required if ≤ 15% of RTP. The power level must be > 15% of RTP to obtain accurate data. At lower power levels, the accuracy of calorimetric data is questionable. The NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric are > 2%. The value of 2% is adequate because this value is assumed in the safety analyses (102%). The frequency of every 24 hours is based on engineering judgment and industry operating experience.

SR 3.3.1.3

Surveillance Requirement 3.3.1.3 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 31 Effective Full Power Days (EFPD) when greater than 15% of RTP. The outputs of the NIS Power Range channels are normalized to the results of the incore detectors. This surveillance is not required when ≤ 15% of RTP. The CHANNEL CALIBRATION consists only of a single point comparison of the incore to excore NIS AFD when > 15% of RTP. The power level must be > 15% of RTP to obtain accurate data. At lower power levels, the accuracy of the data would be questionable. The excore NIS channel shall be recalibrated if the absolute difference between the incore and excore AFD is > 3%. The value of 3% is acceptable because this value is used in the Setpoint Methodology Study (Ref. 5). The frequency of every 31 EFPD is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

Surveillance Requirement 3.3.1.4 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST every 31 days on a STAGGERED TEST BASIS. This test shall verify the OPERABILITY of the RTBs by actuation of the end devices. This test shall include independent verification of the Undervoltage and Shunt Trip Mechanisms. This test must be performed on the bypass breakers prior to their being placed in service. The frequency of every 31 days on a STAGGERED TEST BASIS is justified in Reference 7.

SR 3.3.1.5

Surveillance Requirement 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. The time allowed for the testing, four hours, and the frequency of every 31 days on a STAGGERED TEST BASIS are justified in Reference 7, Table 3.1-1.

SR 3.3.1.6

Surveillance Requirement 3.3.1.6 is the performance of a CHANNEL CALIBRATION for the NIS power range channels every 92 Effective Full Power Days (EFPD) when > 50% of RTP. The outputs of the NIS Power Range channels are normalized to the results of the incore detectors. This surveillance is not required when ≤ 50% of RTP. The CHANNEL CALIBRATION consists of a calibration of the excore NIS channels based on the incore system results. The power level must be > 50% of RTP because lower power levels do not provide results that are as accurate. The time allowed for the testing, four hours, and the surveillance interval of 92 EFPD are justified in Reference 7, Table 3.1-1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

Surveillance Requirement 3.3.1.7 is the performance of an ANALOG CHANNEL OPERATIONAL TEST every 23 days on a STAGGERED TEST BASIS. This test is a periodic check of the analog process control equipment while the unit is at power. When the channel is placed in the test condition, the input to the SSPS is changed to the tripped condition, and the input from the transmitter is removed. This allows a test signal to be introduced into the instrument loop. The input to the bistable can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The Trip Setpoint of the bistable can be determined by varying the input and observing the bistable test lamp. Individual process control modules may be tested in place using multiple sets of test jacks, or by module removal and verification in a calibration laboratory. If individual modules are checked, a verification of the loop accuracy is necessary to satisfy the statistical analyses assumptions. This test is performed every 23 days on a STAGGERED TEST BASIS and is justified in Reference 7, Table 3.1-1.

SR 3.3.1.8

Surveillance Requirement 3.3.1.8 is the performance of an ANALOG CHANNEL OPERATIONAL TEST as described in SR 3.3.1.7, except that the test is performed every 92 days with the unit in MODES 3, 4, or 5 and with the Rod Control System capable of rod withdrawal. This test ensures that the NIS source range channels are OPERABLE prior to taking the reactor critical. This test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition.

SR 3.3.1.9

Surveillance Requirement 3.3.1.9 is the performance of an ANALOG CHANNEL OPERATIONAL TEST as described in SR 3.3.1.7, except that it is performed every 92 days with the unit in MODES 3, 4, or 5. This test ensures that the NIS source range channels are OPERABLE for protection from a boron dilution accident. This test shall include verification of the High Flux at Shutdown alarm setpoint.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

Surveillance Requirement 3.3.1.10 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST as described in SR 3.3.1.4. The test is performed every 31 days on a STAGGERED TEST BASIS and is justified in Reference 7, Table 3.1-1.

SR 3.3.1.11

Surveillance Requirement 3.3.1.11 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 18 months, or approximately every refueling. The 18 month assumption is made in the determination of the magnitude of the transmitter drift in statistical analyses. This test is a complete check of the process control instrument loop and the transmitter. The transmitter "as found" value is noted and adjustments are made as necessary, with notation of the "as left" value for use in a drift calculation. The transmitter may be calibrated in place by use of deadweight or hydraulic/pneumatic testing equipment, on a bench using essentially the same type of equipment, or be replaced by an equivalent unit calibrated in a laboratory. Resistance temperature detector (RTD) channels may be calibrated in place using cross-calibration techniques, in a test bath after removal from the piping, or replaced by a previously calibrated unit. An ANALOG CHANNEL OPERATIONAL TEST is performed on the analog equipment, process control rack modules, with notation of the "as found" and "as left" bistable Trip Setpoints. Completion of this test results in the channel being properly adjusted and expected to remain within the Allowable Value until the next scheduled surveillance.

SR 3.3.1.12

Surveillance Requirement 3.3.1.12 is the performance of a CHANNEL CALIBRATION every 18 months, or about every refueling. The neutron detectors may be excluded from the CHANNEL CALIBRATION. This surveillance is not required for the NIS power range detectors for entry into MODES 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2 because the unit 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 frequency of every 18 months is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

Surveillance Requirement 3.3.1.13 is the performance of an ANALOG CHANNEL OPERATIONAL TEST, as described in SR 3.3.1.7, except that the test is performed every 18 months, or about every refueling. The frequency of 18 months is based on engineering judgment and industry operating experience.

SR 3.3.1.14

Surveillance Requirement 3.3.1.14 is the performance of an ANALOG CHANNEL OPERATIONAL TEST, as described in SR 3.3.1.7, except that the test is performed every 18 months, or about every refueling. This test will verify that the associated interlock is in its required state when the [power level] is $\geq$ the [interlock Trip Setpoint]. The frequency of 18 months is based on engineering judgment and industry operating experience.

SR 3.3.1.15

Surveillance Requirement 3.3.1.15 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST, as described in SR 3.3.1.4, except that the test is performed every 18 months, or about every refueling. The frequency of 18 months is based on engineering judgment and industry operating experience.

SR 3.3.1.16

Surveillance Requirement 3.3.1.16 is the performance of an ANALOG CHANNEL OPERATIONAL TEST, as described in SR 3.3.1.7, except that it is performed prior to reactor startup. This surveillance is only required if it has not been performed within the previous 31 days. Performance of this test prior to reactor startup will ensure that the NIS source, intermediate, and power range channels are OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.17

Surveillance Requirement 3.3.1.17 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST, as described in SR 3.3.1.4, except that the test is performed prior to reactor startup. This surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this surveillance. Performance of this test will ensure that the turbine trip-reactor trip functions are OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.18

This Surveillance Requirement ensures that the train actuation response times are verified on a STAGGERED TEST BASIS. The response time values are provided in [document] and are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment reaches the required functional state, e.g., pumps at rated discharge pressure, valves in full open or closed position. Each train's response must be verified every 18 months on an STAGGERED TEST BASIS ,i.e., Train A at 18 months after initial startup, Train B at 36 months, and then Train A again. The 18 month intervals are based on engineering judgment and industry operating experience. Response times cannot be determined at power, since equipment operation is required. The test may be performed in one measurement or in overlapping segments, with verification that all components are measured.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section [7.3], Engineered Safety Features Actuation System.

2. Watts Bar FSAR, Section [7.2], Reactor Trip System.

3. Watts Bar FSAR, Chapter [15], Accident Analysis.

4. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

5. WCAP-12096, "Westinghouse Setpoint Methodology For Protection Systems, Watts Bar Units 1 and 2", April 1989.

6. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."

7. WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System".

8. Generic Letter 85-09, "Technical Specifications for Generic Letter 83-28, Item 4.3".

9. WCAP-7672, "Solid State Logic Protection System Description".

10. WCAP-7488L, "Solid State Protection System Description".

11. WCAP-10835, "Report of the DS-416 Reactor Trip Breaker Undervoltage and Shunt Trip Attachments Life Cycle Tests".

12. WCAP-10852, "Report of the DB-50 Reactor Trip Breaker Shunt and Undervoltage Trip Attachments Life Cycle Tests".

13. WCAP-10272-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System".

14. NUREG-1217, "Evaluations of Safety Implications of Control Systems in LWR Nuclear Power Plants", 4/88.

15. NUREG-1218, "Regulatory Analysis for Proposed Resolution of USI A-47", 4/88

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Features Actuation System Instrumentation

BASES

BACKGROUND

The primary purpose of the Engineered Safety Features Actuation System (ESFAS) is to initiate necessary safety system actuation, and/or a unit shutdown, based upon the values of selected unit parameters.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below (refer to Figure B 3.3.2-1 for the discussion of the ESFAS):

- field transmitters or process sensors and instrumentation: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

- signal process control and protection system including analog protection system, Nuclear Instrumentation System, field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications.

- solid state protection system (SSPS) including input, logic, and output bays: initiates the proper unit shutdown/ESF actuation in accordance with the defined logic and based upon the bistable outputs from the signal process control and protection system.

The OPERABILITY of ESFAS is necessary to meet the assumptions of the safety analyses and provide for the mitigation, and in some cases, termination, of accident and transient conditions.

Field Instruments and Sensors

In order to meet the design demands for redundancy and reliability, more than one, and often as many as four, field sensors or transmitters are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the setpoint

(continued)

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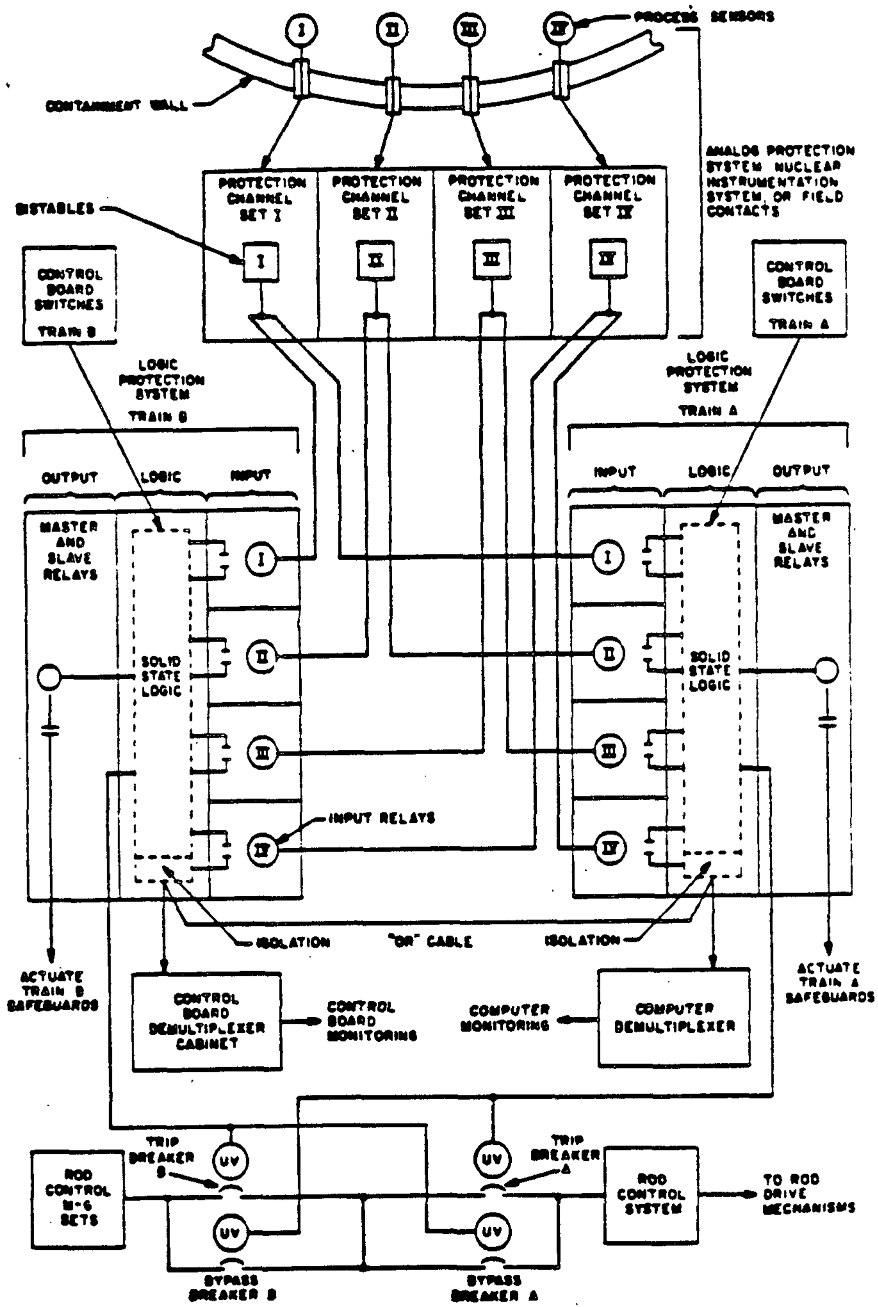

Figure B 3.3.2-1 (Page 1 of 1)

Engineered Safety Features Actuation System

BASES (continued)

BACKGROUND (continued)

Allowable Value. The OPERABILITY of each instrument loop, excluding the transmitter, can be evaluated when its "as found" value is compared against the allowable value noted above.

Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. 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 References 1, 2, and 3. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS while others provide input to the SSPS, main control board, plant computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a 2/3 logic are sufficient to provide the required reliability and redundancy. If one channel fails in the non-conservative direction, the function is still OPERABLE with a 2/2 logic. If one channel fails in the conservative direction, a trip will not occur because of the single failure and the function is still OPERABLE with a 1/2 logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a 2/4 logic are sufficient to provide the required reliability and redundancy. The circuit must be 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. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE 279 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

(continued)

BASES (continued)

BACKGROUND (continued)

The Trip Setpoints noted in Table 3.3.2-1 are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for channel calibration accuracy, i.e., ± (Rack Calibration + Comparator Setting Accuracy).

Each channel of the process control equipment can be tested on line to verify that the signal/setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. Surveillance Requirements for the channels are specified in the Surveillance Requirements section.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based upon the methodology described in the RPS/ESFAS Setpoint Methodology Study (Ref. 5) 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 sensor and signal processing equipment for these channels are expected to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System (SSPS)

The SSPS equipment is used for the decision logic processing of alarm outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. In the event that one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. In the event that both trains are taken out of service or placed in test, a trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements set down by the regulatory guides. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS provides the decision logic which warrants ESF actuation; generates the electrical output signal which will initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

(continued)

BASES (continued)

BACKGROUND (continued)

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices which represent combinations indicative of various transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays, to those components whose aggregate function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the section on Applicable Safety Analyses.

The decision logic matrix functions are described in the FSAR Section [7.2]. In addition to the Reactor Trip/Engineered Safety Features, these figures also describe the various "permissive interlocks" which are associated with unit conditions. Each train has a built-in testing device which can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The slave relay test actuates the end devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the slave relay test circuit and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY

Design Basis Definition

The ESFAS and interlocks are designed to ensure that the following operational criteria are met:

- The associated actuation will occur when the parameter monitored by each channel reaches, or exceeds, its setpoint and the specified coincidence logic is satisfied;
- separation and redundancy is maintained to permit a channel or train to be out of service for testing or maintenance while still maintaining reliability within the ESF actuation instrumentation network;
- functional capability is provided from diverse parameters.

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 Pressure--Low is a primary actuation signal for small LOCAs and a backup actuation signal for Steam Line Breaks outside containment. Note that manual initiation of ESF is not relied upon in the safety analyses.

The ESFAS is part of the primary success path which functions or actuates to mitigate a MODE 1, 2, 3, or 4 design basis accident (DBA) that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement (Ref. 6).

1. SAFETY INJECTION

Safety Injection (SI) provides two primary functions;

1) primary side mass addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal and clad integrity, Peak Clad Temperature $< 2200\ ^{\circ}F$), and,

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

2) boration to ensure recovery and maintenance of shutdown margin (k_{eff} < 1.0)

These functions are necessary to mitigate the effects of high energy line breaks both inside and outside of containment. The SI signal is also used to initiate other functions such as:

- Phase A Containment Isolation,
- Containment Vent Isolation,
- Reactor Trip,
- Turbine Trip,
- Feedwater Isolation,
- Start of Motor-Driven Auxiliary Feedwater Pumps,
- Control Room Ventilation Isolation, and
- provides an enable signal to Automatic Switchover to Containment Sump (Coincident With RWST Level--Low-Low)

These other functions ensure:

- isolation of nonessential systems through containment penetrations,
- trip of the turbine and reactor (to limit power generation),
- isolation of main feedwater (to limit secondary side mass losses),
- starting auxiliary feedwater (to ensure secondary side cooling capability),
- isolation of the Control Room (to ensure habitability), and
- enabling of refueling water storage tank (RWST) switchover on low-low level (to ensure continued cooling via use of the containment sump)

1.A Manual Initiation

The operator can initiate SI at any time by using either of two switches in the Control Room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals. Manual Initiation of SI must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur. This function is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

1.C Containment Pressure--High

This signal provides protection against the following accidents:

- Steam Line Break (inside containment),
- Loss of Coolant Accident, and
- Feed Line Break (inside containment)

The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure--High provides no input to any control functions. Therefore, it satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 channel logic. 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.

Containment Pressure--High must be OPERABLE in MODES 1, 2 and 3, when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. There would be a significant increase of the containment temperature and pressure, thus allowing detection and mitigation. In MODES 4, 5, and 6 there is insufficient energy in the primary or secondary systems to pressurize the containment and allow detection of an accident.

1.D Pressurizer Pressure--Low

This signal provides protection against the following accidents:

- Inadvertent Opening of a Steam Generator Relief or Safety Valve,
- Steam Line Break,
- a Spectrum of Rod Cluster Control Assembly Ejection Accidents (Rod Ejection),
- Inadvertent Opening of a Pressurizer Relief or Safety Valve,
- Loss of Coolant Accidents, and
- Steam Generator Tube Rupture (SGTR)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

Pressurizer Pressure provides both control and protection functions; input to the Pressurizer Pressure Control System, Reactor Trip, and SI. Therefore, the actuation logic must be 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. The function satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 logic. For units that have dedicated protection and control channels, only 3 protection channels are necessary to satisfy the requirements of IEEE 279 (Ref. 4). 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 or Steam Line Break 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 P-11), to mitigate the consequences of a LOCA or stuck open safety or relief valve. This signal may be manually blocked by the operator when 2/3 pressurizer pressure channels indicate below the P-11 setpoint. Automatic SI actuation below this pressure is then performed by the Containment Pressure--High signal.

This function is not required to be OPERABLE in MODE 3 below the P-11 setpoint or in MODE 4. Other ESF functions are used to detect accident conditions and actuate the ESF systems in these MODES. In MODES 4, 5 and 6, accident detection and mitigation is accomplished by the operator.

1.E Steam Line Differential Pressure

Steam Line Differential Pressure provides protection against the following accidents:

- Steam Line Break,
- Feed Line Break, and
- Inadvertent Opening of a Steam Generator Relief or Safety Valve

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

Steam Line Differential Pressure provides no input to any control functions, therefore, it satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 channel logic, on each steam line. With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during a Steam Line Break event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

Steam Line Differential Pressure must be OPERABLE in MODES 1, 2 and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This function is not required to be OPERABLE in MODES 4, 5, or 6 because there is not sufficient energy in the secondary side of the unit to have an accident.

1.F High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low <u>OR</u> Coincident With Steam Line Pressure Low

This function provides protection against the following accidents:

- Steam Line Break, and
- the Inadvertent Opening of a Steam Generator Relief or Safety Valve

[High Steam Flow satisfies the requirements of IEEE 279 (Ref. 4) with 1/2 channel logic, on each steam line in at least 2 steam lines because the control function can not initiate the event.] With the transmitters (d/p cells) typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during a Steam Line Break event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. The Trip Setpoint for High Steam Flow is a linear function that varies with power level. The function is a ΔP corresponding to 40% of full steam flow at 0% load. Then a ΔP increasing linearly to ΔP corresponding to 110% of full steam flow at full load. The Allowable Value is similarly calculated. The T_{avg} signal satisfies the requirements of IEEE 279 (Ref. 4) with 1 channel per

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

loop in two loops because the control channel can not initiate the event. With the transmitters typically located inside the containment, it is possible for them to experience adverse environmental conditions during a LOCA event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. Steam Line Pressure--Low provides no input to any control functions, therefore, it satisfies the requirements of IEEE 279 (Ref. 4) with 1 channel, on each steam line. With the transmitters typically located inside the steam tunnels, 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.

This function must be OPERABLE in MODES 1, 2 and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This signal may be manually blocked by the operator when below the P-12 setpoint. Above P-12, this function is automatically unblocked. Below P-12, inside containment Steam Line Break will be terminated by automatic SI actuation via Containment Pressure--High, and outside containment Steam Line Break will be terminated by the [Steam Line Differential Pressure signal. This function is not blocked for steam line isolation.] This function is not required to be OPERABLE in MODES 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have an accident.

2. <u>CONTAINMENT SPRAY</u>

Containment Spray provides 3 primary functions;

- 1) lower containment pressure and temperature after a high energy line break in containment, and
- 2) reduce the amount of radioactive iodine in the containment atmosphere, and
- 3) adjust the pH of the water in the containment recirculation sump after a large break LOCA.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

These functions are necessary to ensure;

- The pressure boundary integrity of the containment structure,
- To limit the release of radioactive iodine to the environment in the event of a failure of the containment structure, and
- To minimize corrosion of the components and systems inside containment following a LOCA.

Actuation of Containment Spray starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps. When the RWST is empty, the spray pump suctions are shifted to the containment sump if continued Containment Spray is required. Containment Spray is actuated manually or by Containment Pressure--High-High.

2.A Manual Initiation

The operator can initiate Containment Spray at any time from the Control Room by turning the Containment Spray actuation switches. Because an inadvertent actuation of Containment Spray could have such serious consequences, two switches must be turned simultaneously to initiate Containment Spray. There are two sets of two switches each in the Control Room. Simultaneously turning the two switches in either set will actuate Containment Spray in the same manner as the automatic actuation signal. Note that Manual Initiation of Containment Spray also actuates Phase "B" Containment Isolation and Containment Vent Isolation.

Manual Initiation of Containment Spray must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond to mitigate the consequences of abnormal conditions.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

2.C Containment Pressure--High-High

This signal provides protection against a LOCA or a Steam Line Break inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure--High-High provides no input to any control functions, and actuates Containment Spray, Main Steam Isolation, and Phase B Containment Isolation. Therefore, it satisfies the requirements of IEEE 279 (Ref. 4) with 2/3 channel logic. However, for enhanced reliability, this function has been designed with 2/4 logic. This is one of the only functions which require the bistable output to energize to perform its required action. It is not desired to have a loss of power actuate Containment Spray since the consequences of an inadvertent actuation of Containment Spray could be serious. Since the transmitters and electronics are located outside of containment, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure--High-High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. There would be a significant increase in the containment temperature and pressure, thus allowing detection and mitigation of the accident. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure--High-High setpoint.

3. <u>CONTAINMENT ISOLATION</u>

Containment Isolation provides isolation of the containment atmosphere, and all process systems which penetrate containment, from the environment. This function is necessary to prevent or limit the release of fission product radioactivity to the environment in the event of a large break LOCA. Containment Isolation is separated into two categories. The first category isolates all process systems penetrating containment except for component cooling water. This is accomplished by Phase "A" Containment Isolation. The second category isolates the component cooling water lines penetrating containment. This is accomplished by Phase "B" Containment Isolation.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

3.A Phase A Containment Isolation

Phase A Containment Isolation is actuated by Safety Injection, discussed previously, or manually. All process lines penetrating containment with the exception of component cooling water are isolated. Component cooling water is not isolated at this time to permit continued operation of the reactor coolant pumps with cooling water flow to the thermal barrier heat exchangers and air/oil coolers. All process lines not equipped with remote operated isolation valves are manually closed or otherwise isolated prior to reaching MODE 4. Manual Phase "A" Containment Isolation is accomplished by either of two [switches] in the Control Room. Note that manual actuation of Phase "A" Containment Isolation also actuates Containment Vent Isolation.

Manual Initiation of Phase "A" Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4 when 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 Phase "A" Containment Isolation. There will also be adequate time for the operator to evaluate unit conditions and manually actuate pumps, systems, and other equipment in response to abnormal or accident conditions.

3.B Phase B Containment Isolation

Phase B Containment Isolation is actuated by Containment Pressure--High-High, discussed previously, or manually. Component cooling water lines penetrating containment are isolated. For containment pressure to reach a value high enough to actuate Containment Pressure--High-High, a large break LOCA or Steam Line Break must have occurred and Containment Spray must have been actuated. Reactor coolant pump operation will no longer be required and component cooling water to the RCPs is, therefore, no longer necessary. The RCPs are manually shut down on a Phase B isolation. Manual Phase "B" Containment Isolation is accomplished by the same [switches] that actuate Containment Spray. When the two [switches] in either set are turned simultaneously, Phase "B" Containment Isolation and Containment Spray will be actuated.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

Manual Initiation of Phase "B" Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4 when 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 Phase "B" Containment Isolation. There will also be adequate time for the operator to evaluate unit conditions and manually actuate pumps, systems, and other equipment in response to abnormal or accident conditions.

3.C Containment Vent Isolation

Containment Vent Isolation closes the containment isolation valves in the purge system. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The purge system has inner and outer containment isolation valves in their supply and exhaust ducts. Containment Vent Isolation is actuated by a Safety Injection Signal, by manual initiation of Phase A or Phase B Containment Isolation, by high radioactivity levels in the purge exhaust, and by high radioactivity levels in the containment atmosphere.

Manual Initiation of Containment Vent Isolation must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur. The requirements for Containment Vent Isolation in MODE 6 when handling fuel are contained in LCO 3.9.4, Containment Building Penetrations.

Safety Injection also initiates Containment Vent Isolation. Safety Injection was discussed previously. A manual Phase "A" or Phase "B" Containment Isolation also actuates Containment Purge Isolation. Phase "A" and Phase "B" Containment Isolation were discussed previously.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY

4. STEAM LINE ISOLATION

Isolation of the main steam lines provides protection in the event of a Steam Line Break inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one steam generator at most. For a Steam Line Break upstream of the isolation valves, inside or outside of containment, closure of the isolation valves will limit the accident to the blowdown from only the effected steam generator. For a Steam Line Break downstream of the isolation valves, closure of the isolation valves will terminate 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 auxiliary feedwater pump during a Feed Line Break.

4.A Manual Initiation

Manual initiation of steam line isolation can be accomplished from the Control Room by the operator. There are two switches in the Control Room. Turning either switch will cause immediate closure of all Main Steam Isolation Valves.

Manual Initiation of Steam Line Isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and steam generators to have a Steam Line Break or other accident resulting in the release of significant quantities of energy to cause a cooldown of the primary system. In MODES 4, 5, and 6, there is insufficient energy in the RCS and steam generators to experience a Steam Line Break or other accident releasing significant quantities of energy.

4.C Containment Pressure--High-High

This function actuates closure of the main steam isolation valves in the event of a LOCA or a Steam Line Break inside containment to maintain at least two unfaulted steam generators as a heat sink for the reactor and to limit the mass/energy release to containment. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure--High-High provides no

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

4.C (continued)

input to any control functions. Therefore, it satisfies the requirements of IEEE 279 (Ref. 4) with 2/4 channel logic. The transmitters and electronics are located outside containment, thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure--High-High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. There would be a significant increase in the containment pressure, thus allowing detection and closure of the main steam isolation valves. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure--High-High setpoint.

4.D High Steam Flow in Two Steam Lines Coincident With Steam Line Pressure--Low <u>OR</u> Coincident With T_{avg}--Low-Low

This function provides closure of the main steam isolation valves during a Steam Line Break or Inadvertent Opening of a Steam Generator Relief or Safety Valve to maintain at least two unfaulted steam generator as a heat sink for the reactor and to limit the mass/energy release to containment. High Steam Flow satisfies the requirements of IEEE 279 (Ref. 4) with 1/2 channel logic, on each steam line in at least two steam lines. With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during a Steam Line Break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. The Trip Setpoint for High Steam Flow is a linear function that varies with power level. The function is a ΔP corresponding to 40% of full steam flow at 0% load. Then a ΔP increasing linearly to a ΔP corresponding to 110% of full steam flow at full load. The Allowable Value is similarly calculated. The Steam Line Pressure--Low signal was discussed previously. The T_{avg} signal satisfies the requirements

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

of IEEE 279 (Ref. 4) with 1 channel per loop in any two loops. With the transmitters typically located inside the containment, it is possible for them to experience adverse environmental conditions during a LOCA event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This function must be OPERABLE in MODES 1, 2 and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. 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 unit to have an accident.

5. <u>TURBINE TRIP AND FEEDWATER ISOLATION</u>

The primary functions of the Turbine Trip and Feedwater Isolation are to prevent damage to the turbine due to water in the steam lines and to stop the excessive flow of feedwater into the steam generators. These functions are necessary to mitigate the effects of a high water level in the steam generators which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The steam generator high water level is due to excessive feedwater flows. This could result in excessive cooldown of the reactor coolant or overflowing the steam generator and filling the steam lines which are not qualified for dead weight.

This function is actuated by Steam Generator Water Level--High-High or by an SI signal. In the event of Safety Injection, the unit is taken off line and the turbine generator must be tripped. The Main Feedwater System is also taken out of operation and the Auxiliary Feedwater System is automatically started. The SI signal was discussed previously.

5.B Steam Generator Water Level--High-High

This signal provides protection against excessive feedwater flow. High water level in the steam generator will result in a loss of moisture removal capability and the carryover of water into the steam lines. Excessive moisture content in the steam will cause damage to the turbine. Other considerations of high water level in the steam generator are an excessive cooldown of the RCS and overflowing the steam generator filling the steam lines not qualified for that amount of dead weight. The transmitters (d/p cells)

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

are located inside containment and also provide input to the Steam Generator Water Level Control System. Therefore, the actuation logic must be 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. This results in a 2/3 actuation logic. For units that have only 3 channels, justification is provided in References 12 and 13. The Trip Setpoint reflects only steady state instrument uncertainties.

This function must be OPERABLE in MODES 1, 2, and 3 when the Main Feedwater System is in operation and in MODES 1 and 2 the turbine generator may be in operation. In MODES 4, 5, and 6, the Main Feedwater System and the turbine generator are not in service and this function is not required to be OPERABLE.

6. <u>AUXILIARY FEEDWATER</u>

The Auxiliary Feedwater System (AFS) is designed to provide a secondary side heat sink for the reactor in the event that the Main Feedwater System is not available. The system has two motor-driven pumps and a turbine-driven pump making it available during normal unit operation, during a loss of AC power, during a loss of main feedwater, and during a feedwater system pipe break. The normal source of water for the AFS is the condensate storage tank (nonsafety-related). A low level in the condensate storage tank will automatically realign the pump suctions to the Essential Raw Cooling Water System (safety-related). The AFS is aligned so that upon a pump start, flow is initiated to the respective steam generators immediately.

6.B Steam Generator Water Level--Low-Low

Steam Generator Water Level--Low-Low provides protection against a loss of heat sink. A feedwater system pipe break, inside or outside of containment, or a loss of main feedwater would result in a loss of steam generator water level. Steam Generator Water Level--Low-Low provides input to the Steam Generator Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protective function actuation) and a single failure in the other channels providing the protection function actuation. This results in a 2/3 actuation logic. For units that have only 3 channels, justification is provided by having a

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

reactor trip on Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch. 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.

This function must be OPERABLE in MODES 1, 2, and 3 to ensure that the steam generators remain the heat sink for the reactor. Steam Generator Water Level--Low-Low in any operating steam generator will cause the motor-driven auxiliary feedwater pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the steam generators. Steam Generator Water Level--Low-Low in any two operating steam generators will cause the turbine-driven pump to start. This function does not have to be OPERABLE in MODES 4, 5, and 6 because there is not enough heat being generated in the reactor to require the steam generators as a heat sink.

6.C Safety Injection Signal

The actuation of an SI signal (discussed previously) automatically starts the motor-driven and turbine-driven auxiliary feedwater pumps. This function must be OPERABLE in MODES 1, 2, and 3. This ensures that at least two steam generators are provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. This function does not have to be OPERABLE in MODES 4, 5, and 6 because there is not enough heat being generated in the reactor to require the steam generators as a heat sink.

6.D Loss Of Offsite Power

A loss of offsite power provides indication of a loss of all AC power. A loss of offsite power to the [service] busses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each [service] bus. A loss of power to either [service] bus will start the turbine-driven and motor-driven auxiliary feedwater pumps to ensure that at least two steam generators contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. This function must be OPERABLE in MODES 1, 2, and 3.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

This function does not have to be OPERABLE in MODES 4, 5, and 6 because there is not enough heat being generated in the reactor to require the steam generators as a heat sink.

6.E Trip Of All Main Feedwater Pumps

A Trip Of All Main Feedwater Pumps is an indication of a loss of main feedwater and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no-load temperature and pressure. Each turbine-driven main feedwater pump is equipped with two pressure switches on the control air line for the speed control system. A low pressure signal from either of these pressure switches indicates a trip of that pump. A Trip Of All Main Feedwater Pumps will start the motor-driven and turbine-driven auxiliary feedwater pumps to insure that at least two steam generators are available with water to act as the heat sink for the reactor.

This function must be OPERABLE in MODES 1, 2, and 3 when the main feedwater pumps will be in use. This function does not have to be OPERABLE in MODES 4, 5, or 6 because the main feedwater pumps will not be in use.

6.F Auxiliary Feedwater Pump Suction Pressure--Low

A low pressure on the auxiliary feedwater pump suction provides protection against a loss of the normal supply of water for the pumps, the Condensate Storage Tank (CST). The loss of water supply may be due to a low level in the CST or due to a CST/pipe rupture for non-seismic qualified CSTs. Three pressure switches are located on each auxiliary feedwater pump suction line from the CST. A low pressure sensed by any two of the switches will cause the emergency supply of water for the pumps to be aligned. Essential Raw Cooling Water (safety-grade) is then lined up to supply the auxiliary feedwater pumps to ensure an adequate supply of water for the Auxiliary Feedwater System (AFS) to maintain at least two of the steam generators as the heat sink for reactor decay heat and sensible heat removal. Since the detectors are located outside containment, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

This function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety-grade supply of water for the AFS to maintain the steam generators as the heat sink for the reactor. This function does not have to be OPERABLE in MODES 4, 5, and 6 because there is not enough heat being generated in the reactor to require the steam generators as a heat sink.

7. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head RHR pumps draw the water from the containment recirculation sump, pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. The suctions of the low head RHR pumps are switched from the RWST to the containment sump on the RWST Level-Low--Low Coincident With an SI signal AND Coincident With Containment Sump Level--High. (The SI signal was discussed previously.) This action must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability.

7.B RWST Level--Low-Low Coincident With Containment Sump Level--High AND Coincident With Safety Injection

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. As the RWST empties, the water from the RWST and the water from the RCS and accumulators will accumulate inside containment via the break. To ensure that there is sufficient water available in containment to support the recirculation phase of the accident, a Containment Sump Level--High signal must be present, in addition to the SI signal and the RWST Level--Low-Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with 4 level transmitters. These transmitters provide no control functions, therefore, a 2/4 logic is adequate to initiate the protective function actuation. Although only 3 channels would be sufficient (Ref. 4), a fourth channel has been added for increased reliability. The transmitters are located inside containment, and thus possibly experience adverse environmental conditions. Therefore, the Trip Setpoint reflects the inclusion of both steady state and environmental instrument uncertainties.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

Containment Sump Level--High must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the RHR pumps prior to emptying the RWST and causing damage to the RHR pumps. This function is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting 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 unit systems.

8. CONTROL ROOM EMERGENCY VENTILATION

The control room must be kept habitable for the operators stationed there during accident recovery and post-accident operations. The control room is equipped with its own ventilation system to provide a habitable environment for the operators. During an accident, the control room is isolated and the Control Room Emergency Ventilation System is put into operation. Control room isolation consists of automatically positioning the appropriate dampers to isolate unfiltered outside air from the Control Room. Control Room Emergency Ventilation is initiated:

- Manually,
- By Phase "A" Containment Isolation, and
- By high gaseous radioactivity in the control room air intakes

Control Room Emergency Ventilation must be OPERABLE in all MODES to ensure a habitable environment for the control room operators.

8.A Manual Initiation

The control room operator can manually initiate Control Room Emergency Ventilation at any time by turning either of two switches in the.control room.

8.C Phase A Containment Isolation

Phase A Containment Isolation actuates Control Room Emergency Ventilation. Phase A Containment Isolation was discussed previously.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)	8.D Area Radiation The Control Room air intakes are continuously monitored by redundant radiation monitors. A High radiation signal from either monitor will initiate Control Room isolation. 9. <u>LOSS OF POWER</u> To be provided.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs AND APPLICABILITY (continued)

10. ESFAS INTERLOCKS

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur.

10.A Reactor Trip--P-4

The P-4 interlock is enabled when either RTB and its associated bypass breaker are open. If this condition is not satisfied, automatic actuation of SI can not occur until the RTBs have been manually reset. The functions of the P-4 interlock are:

- Trip the main turbine,
- Isolate main feedwater with coincident low T_{avg},
- Prevent reactuation of safety injection after a manual reset,
- Transfer the steam dump from the load rejection controller to the plant trip controller, and
- Prevent opening of the main feedwater isolation valves if they were closed on SI or Steam Generator Water Level--High-High

This function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This function does not have to be OPERABLE in MODES 4, 5, or 6 because the main turbine, the Main Feedwater System, and the Steam Dump System are not in operation.

10.B Pressurizer Pressure--P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of an SI. With 2/3 pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure--Low SI signal (previously discussed). With 2/3 pressurizer pressure channels $\geq$ P-11 setpoint, the Pressurizer Pressure--Low SI signal is automatically enabled. The operator can also enable the trip by use of the respective manual reset buttons. The Trip Setpoint reflects only steady state instrument uncertainties. This function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of a SI. This function does not have to be OPERABLE in MODES 4, 5, or 6 because unit pressure must already be below the P-11 setpoint for the requirements of the heatup/cooldown curves to be met.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)

10.C T_{avg}--Low-Low, P-12

On increasing reactor coolant temperature, the P-12 interlock reinstates SI on High Steam Flow Coincident With Steam Line Pressure--Low OR Coincident With T_{avg}--Low-Low. On an increasing temperature, the P-12 interlock also provides an arming signal to the Steam Dump System. On decreasing reactor coolant temperature, the P-12 interlock allows the operator to manually block SI on High Steam Flow Coincident With Steam Line Pressure--Low OR Coincident With T_{avg}--Low-Low. On a decreasing temperature, the P-12 interlock also removes the arming signal to the Steam Dump System to prevent an excessive cooldown of the RCS due to a malfunctioning Steam Dump System.

This function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This function does not have to be OPERABLE in MODES 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have an accident.

10.D Steam Generator Water Level--High-High, P-14

The P-14 interlock is actuated when level in any steam generator exceeds the high-high setpoint and performs the following functions:

- Trips the main turbine,
- Trips the main feedwater pumps,
- Initiates Feedwater Isolation, and
- Shuts the main feedwater regulating valves and the bypass feedwater regulating valves

The main feedwater pumps are tripped, Feedwater Isolation is actuated, and the main and bypass feedwater regulating valves are closed to prevent any further addition of water to the steam generators. The main turbine is tripped to prevent carryover of excessive moisture to the turbine which would damage the turbine.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCOs, AND APPLICABILITY (continued)	This function must be OPERABLE in MODES 1 and 2 when the turbine generator may be in operation and in MODES 1, 2, and 3 when the Main Feedwater System may be in operation. This function does not have to be OPERABLE in MODES 4, 5, or 6 because the turbine generator and the Main Feedwater System are not in service.
ACTIONS	In the event a channel's Trip Setpoint is found non-conservative with respect to the Allowable Value; or the transmitter, a rack module, or an SSPS module is found inoperable, then the function which that channel provides must be declared inoperable and the unit must enter the Condition statement for the particular protection function affected. If the number of inoperable channels or trains for a particular protection function is greater than the number of inoperable channels addressed by the Condition statement, then the unit no longer meets the assumptions of the safety analyses and must be placed in a MODE where the function is no longer required to be OPERABLE as per LCO 3.0.3.

Condition A

Condition A is applicable to all ESFAS protection functions. The most frequent occurrence to render a protection function inoperable is the determination that a bistable or process module has drifted sufficiently to exceed the Allowable Value. Typically, the drift is not large, which would result in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of an ANALOG CHANNEL OPERATIONAL TEST, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the Allowable Value, the channel must be declared inoperable immediately and the appropriate Condition from Table 3.3.2-1 must be entered. Condition A addresses the situation where one or more channels for one or more functions are inoperable at the same time. 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)

BASES (continued)

ACTIONS (continued)

Condition B

Condition B is applicable to manual initiation of:

- Safety Injection,
- Containment Spray,
- Phase "A" Containment Isolation,
- Phase "B" Containment Isolation,
- Trip of All Main Feedwater Pumps, and
- Auxiliary Feedwater Pump Suction Pressure--Low

This action addresses the train orientation of the SSPS for the first four functions. For the last two functions, the OPERABILITY of the AFS must be assured by allowing automatic start of the AFS pumps and a continued supply of safety-grade water (ERCW). If a train or channel is inoperable, 48 hours is allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours is allowed to place the unit in MODE 3. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action B.2.1, an additional 30 hours is a reasonable time, based on industry operating experience and normal cooldown rates, to reach MODE 5, where the function is no longer required, from MODE 3 without challenging unit systems or operators. In
this MODE, the unit does not have any analyzed transients or conditions which require the explicit use of the protection functions noted above. The allowance of 48 hours to return the train to an OPERABLE status, 6 hours to reach MODE 3, and 30 additional hours to reach MODE 5 are justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition C

Condition C is applicable to the Automatic Actuation Logic and Actuation Relays for the following functions;

- Safety Injection,
- Containment Spray,
- Phase "A" Containment Isolation,
- Phase "B" Containment Isolation, and
- Automatic Switchover to Containment Sump

(continued)

BASES (continued)

ACTIONS (continued)

This action addresses the train orientation of the SSPS and the Master and Slave Relays. If one train is inoperable, 6 hours is allowed for restoration to OPERABLE status. Subsequently, the action required is to perform an orderly shutdown with the unit in MODE 3 within the following 6 hours. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action C.1, an additional 30 hours is a reasonable time, based on industry operating experience and normal cooldown rates, to reach MODE 5, where these functions are no longer required, from MODE 3 without challenging unit systems or operators. Placing the unit in MODE 5 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions which require the explicit use of the protection functions noted above. The allowance of 12 hours to reach MODE 3 and the subsequent 30 hours allowed to reach MODE 5 are justified in Reference 7, evaluated as Maintenance Time in Table 3.1-1.

Condition D

Condition D is applicable to:

- Containment Pressure--High,
- Pressurizer Pressure--Low,
- Steam Line Differential Pressure,
- High Steam Flow in Two Steam Lines Coincident With T_{avg}--Low-Low, OR Coincident With Steam Line Pressure--Low,
- Steam Generator Water Level--High-High,
- Steam Generator Water Level--Low-Low, and
- Loss or Degraded Voltage
- Containment Pressure--High-High,
- RWST Level--Low-Low and Coincident with SI AND Containment Sump Level--High

Each of the above signals meets the requirements of IEEE 279 as per previous discussions. Placing a known failed channel in the tripped condition within 6 hours is adequate. The function is then in a partial trip condition where 1/2 or 1/3 logic will result in actuation. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for a second channel to be in the bypassed condition for testing is justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these time intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

(continued)

BASES (continued)

ACTIONS (continued)

Condition E

Condition E is applicable to the Containment Vent Isolation function. This action addresses the train orientation of the SSPS and the Master and Slave Relays for this function. If a train is inoperable, operation may continue as long as the Containment Purge System supply and exhaust valves are placed and maintained in the closed position within 4 hours. The 4 hour time limit is justified in Reference 7. This is acceptable since the purpose of the actuation function is to close these valves if they are open. Once the valves are closed, this condition may continue for an indefinite period of time.

Condition F

Condition F is applicable to the Containment Vent Isolation Function. This action addresses the containment atmosphere gaseous and particulate radiation monitors. A Completion Time of 4 hours is allowed to adjust setpoints or close the purge and exhaust isolation valves.

Condition G

Condition G is applicable to the Automatic Actuation Logic and Actuation Relays for the Steam Line Isolation, Turbine Trip and Feedwater Isolation and Auxiliary Feedwater Actuation functions. This action addresses the train orientation of the SSPS and the Master and Slave Relays for these functions. If one train is inoperable, 6 hours may be taken for restoration to an OPERABLE status. Subsequently, the action required is to perform an orderly shutdown with the unit in MODE 3 within the following 6 hours. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action G.1, an additional 6 hours is a reasonable time, based on industry operating experience and normal cooldown rates, to reach MODE 4, where these functions are no longer required, from MODE 3 without challenging unit systems or operators. These functions are no longer required in MODE 4. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions which require the explicit use of the protection functions noted above. The allowance of 12 hours to reach MODE 3 and the subsequent 6 hours allowed to reach MODE 4 are justified in Reference 7, evaluated as Maintenance Time in Table 3.1-1.

(continued)

BASES (continued)

ACTIONS (continued)

Condition H

Condition H is applicable to

- Manual initiation of system Steam Line Isolation,
- Loss of Offsite Power, and
- P-4 Interlock

This action addresses the train orientation of the SSPS for these functions. If a train is inoperable, 48 hours is allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours is allowed to place the unit in MODE 3. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is reasonable based on operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action H.2.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where these functions are no longer required, from MODE 3 without challenging unit systems or operators. These functions are no longer required in MODE 4. In this MODE, the unit does not have any analyzed transients or conditions which require the explicit use of the protection functions noted above. The allowance of 48 hours to return the train to an OPERABLE status, 6 hours to reach MODE 3, and 6 additional hours to reach MODE 4 are justified in Reference 7, evaluated as Maintenance Time and Test Time (respectively) in Table 3.1-1. Allowance of these intervals does not result in a significant increase in the probability of a core melt over the expected life of the unit.

Condition I

Condition I is applicable to manual, automatic, and Phase "A" Isolation actuation of Control Room Emergency Ventilation. This action addresses the train orientation of the SSPS for this function. If one train is inoperable, 7 days (168 hours) is permitted to restore the function to an OPERABLE status. If the function cannot be restored to an OPERABLE status, the Control Room Ventilation System must be manually placed in the isolation mode of operation.
This is the mode of operation in which the Control Room Ventilation System is placed upon receipt of a Control Room Isolation signal.

(continued)

BASES (continued)

ACTIONS (continued)

Condition J

Condition J is applicable to the High Radiation Actuation of the Control Room Emergency Ventilation System. With one channel inoperable, the system must be placed in the isolation mode of operation within 1 hour.

Condition K

Condition K is applicable to the P-11, P-12, and P-14 Interlocks. With pressurizer pressure above the P-11 setpoint, the Pressurizer Pressure--Low signal is automatically enabled. With T_{avg} above the P-12 setpoint, SI on High Steam Flow Coincident With either Steam Line pressure--Low or T_{avg}--Low-Low is automatically reinstated. Above the P-12 setpoint, the Steam Dump System arming signal is also reinstated. With two channels inoperable, the operator must verify that the interlock is in the required state for the existing unit conditions. This determination must be made within 1 hour. The Completion Time of 1 hour is based on industry operating experience.

SURVEILLANCE REQUIREMENTS

The Surveillance Requirements for any particular ESFAS function are found in the Surveillance Requirements column of Table 3.3.2-1, for that function.

Note that each channel of process protection supplies both trains of the ESFAS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TESTS are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies. For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate Technical Specification response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1

Surveillance Requirement 3.3.2.1 is the performance of a CHANNEL CHECK. A CHANNEL CHECK is simply the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the 3 (or 4) channels indicated in the control room should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the racks have drifted outside of their limit. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their Trip Setpoints. The surveillance interval, about once every shift, is based on engineering judgment.

SR 3.3.2.2

Surveillance Requirement 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The CONTINUITY CHECK does not have to be performed for this surveillance. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference 7, Table 3.1-1.

SR 3.3.2.3

Surveillance Requirement 3.3.2.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference 7, Table 3.1-1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.4

Surveillance Requirement 3.3.2.4 is the performance of an ANALOG CHANNEL OPERATIONAL TEST. This test is a periodic check of the analog process control equipment while the unit is at power. When the channel is placed in the test condition, the input to the SSPS is changed to the tripped condition and the input from the transmitter is removed. This allows a test signal to be introduced into the instrument loop. The input to the bistable can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The Trip Setpoint of the bistable can be determined by varying the input and observing the bistable test lamp. Individual process control modules may be tested in place using multiple sets of test jacks, or by module removal and verification in a calibration laboratory. If individual modules are checked, a verification of the loop accuracy is necessary to satisfy the statistical analyses assumptions. This test is performed every 92 days and is justified in Reference 7, Table 3.1-1.

SR 3.3.2.5

Surveillance Requirement 3.3.2.5 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is a check of the manual actuation functions and is performed on a quarterly basis (every 92 days). 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 test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. The frequency is justified in Reference 7.

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.6

Surveillance Requirement 3.3.2.6 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 18 months, or approximately every refueling. The 18 month assumption is made in the determination of the magnitude of the transmitter drift in the statistical analyses. This test is a complete check of the process control instrument loop and the transmitter. The transmitter "as found" value is noted and adjustments are made as necessary, with notation of the "as left" value for use in a drift calculation. The transmitter may be calibrated in place by use of deadweight or hydraulic testing equipment, on a bench using essentially the same type of equipment, or be replaced by an equivalent unit calibrated in a laboratory. Resistance temperature detector channels may be calibrated in place using cross-calibration techniques, in a test bath after removal from the piping, or replaced by a previously calibrated unit. An ANALOG CHANNEL OPERATIONAL TEST is performed on the analog equipment (process control rack modules), with notation of the "as found" and "as left" bistable Trip Setpoints. Completion of this test results in the channel being properly adjusted and expected to remain within the Allowable Value until the next scheduled surveillance.

SR 3.3.2.7

Surveillance Requirement 3.3.2.7 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST as in SR 3.3.2.5, except that it is performed every 18 months rather than every 92 days (quarterly).

SR 3.3.2.8

This Surveillance Requirement ensures that the train actuation response times are verified on a STAGGERED TEST BASIS. The response time values are provided in [document] and are the maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment reaches the required functional state, e.g., pumps

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BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8 (continued)

at rated discharge pressure, valves in full open or closed position. Each train's response must be verified every 18 months on an STAGGERED TEST BASIS, i.e., Train A at 18 months after initial startup, Train B at 36 months, and then Train A again. The 18 month intervals are based on engineering judgment and industry operating experience. Response times cannot be determined at power, since equipment operation is required. The test may be performed in one measurement or in overlapping segments, with verification that all components are measured.

REFERENCES

1. Watts Bar FSAR, Section [7.3], Engineered Safety Features Actuation System.
2. Watts Bar FSAR, Section [7.2], Reactor Trip System.
3. Watts Bar FSAR, Chapter [15], Accident Analysis.
4. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations", April 5, 1972.
5. WCAP-12096, "Westinghouse Setpoint Methodology For Protection Systems, Watts Bar Units 1 and 2," April 1989.
6. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."
7. WCAP-10272, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System".
8. WCAP-7672, "Solid State Logic Protection System Description".
9. WCAP-7488L, "Solid State Protection System Description".
10. WCAP-10835, "Report of the DS-416 Reactor Trip Breaker Undervoltage and Shunt Trip Attachments Life Cycle Tests".

(continued)

BASES (continued)

REFERENCES (continued)

11. WCAP-10852, "Report of the DB-50 Reactor Trip Breaker Shunt and Undervoltage Trip Attachments Life Cycle Tests".

12. WCAP-10272-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System".

13. NUREG-1217, "Evaluations of Safety Implications of Control Systems in LWR Nuclear Power Plants", 4/88

14. NUREG-1218, "Regulatory Analysis for Proposed Resolution of USI A-47", 4/88

B 3.3 INSTRUMENTATION

B 3.3.3 Accident Monitoring Instrumentation

BASES

BACKGROUND

The primary purpose of the accident monitoring instrumentation is to display plant variables that provide information required by the control room operators. This information is necessary to perform the manual actions, including long-term recovery actions for which no automatic control is provided, specified in the plant Emergency Operating Procedures (EOPs) associated with design basis events. These variables are designated as Type A, Category 1 in accordance with Reference 1. Type A, Category 1 design and qualification requires seismic and environmental qualification, the application of single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display (Ref. 2). These plant parameters are defined by the plant functional requirements, process block diagrams, specification sheets, and elementary wiring diagrams.

The accident monitoring instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation provides a measurable electronic signal based upon the physical characteristics of the parameter being measured.

- Signal process control provides signal conditioning and a compatible electrical signal output to control board/control room/miscellaneous display devices.

- Display devices provide concise and accurate display of designated plant parameters for the period following an accident and/or seismic event.

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 following an accident. This capability is consistent with the recommendations of References 2 and 3.

(continued)

BASES (continued)

BACKGROUND (continued)

Field Transmitters and Instrumentation

In order to meet the design demands for redundancy and reliability, more than one, and often as many as three, field sensors or transmitters are used to measure plant parameters. The Required Number of Channels for each plant parameter is listed in Table 3.3.3-1.

Transmitters are not usually tested while the plant is on line, and are normally subjected to maintenance only when recalibration is required during a plant shutdown, or when a failed instrument must be replaced. Prior to, and after installation, each field instrument is calibrated to provide the correct measured value and span. Surveillance requirements are discussed in the Surveillance Requirements section.

Signal Process Control

The signal process control equipment provides signal conditioning and compatible output signals for instruments located on the main control board or in the main control room. The signal processing hardware is designed and configured to provide the required channel redundancy, channel separation, and channel independence.

Each channel of the process control equipment can be tested on-line to verify OPERABILITY. Surveillance requirements are discussed in the Surveillance Requirements section.

Display Devices

The operator has immediate access to the display devices which provide him information from redundant or diverse channels in units familiar to the operator, i.e., temperature in °F, not voltage. When two or more instruments are needed to cover a particular range, overlapping of instrument ranges is provided.

APPLICABLE SAFETY ANALYSES

Design Basis Definition

The Accident Monitoring Instrumentation LCO ensures the operability of Type A, Category 1 variables so that the

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

control room operating staff can:

- Perform the diagnosis specified in the EOPs. These variables are restricted to pre-planned actions for design basis events, specifically, LOCA, Steam Line Break, Feedwater Line Break, and Steam Generator Tube Rupture (SGTR), and,

- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety function, and

- Reach and maintain a safe shutdown condition.

No single failure within either the accident monitoring instrumentation, its auxiliary supporting features, or its power sources, concurrent with the failures that are a condition of, or result from, a specific design basis event, shall prevent the operator from being presented the required information. Where failure of one accident monitoring channel results in information ambiguity, e.g., the redundant displays disagree, additional information is provided to allow the operator to deduce the plant's actual condition. This is accomplished by providing additional independent channels of the same variable or by providing diverse channel(s) of another variable. Redundant or diverse channels are electrically independent and physically separated from each other, to the extent practicable with two train separation, and from equipment not classified important to safety in accordance with 10 CFR Part 50 Appendix A, General Design Criteria (GDC) 22 and 24 (Ref. 5), and derived regulatory guides consistent with the licensing basis of the plant.

In the event that diversity is employed in lieu of redundancy, detailed procedures are maintained to detect and resolve any ambiguity that may exist. These procedures recognize such factors as independent and physical separation of the channels employed.

<u>Selection Criteria</u>

The accident monitoring instrumentation is part of the primary success path which functions or actuates to mitigate a MODE 1, 2, or 3 DBA that either assumes the failure of or

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criteria 3 of the NRC Interim Policy Statement (Ref. 4).

LCO

The Accident Monitoring Instrumentation LCO provides the requirements of Type A, Category 1 monitors which provide information required by the control room operators to perform the manual actions specified in the plant EOPs. Table 3.3.3-1 is for illustration purposes only. It does not attempt to encompass every Type A, Category 1 variable at every plant. The table contains the types of variables commonly found and includes variables that utilize each of the prescribed Required Actions. Plant-specific lists will be developed when the specification is made plant-specific. Specific instrument requirements and EOP functions for Table 3.3.3-1 are provided below:

1. Pressurizer Water Level

Pressurizer water level is used to determine whether to terminate Safety Injection, if still in progress, or to re-initiate Safety Injection if it has been stopped. Knowledge of pressurizer water level 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.

2. Reactor Coolant Pressure (Wide Range)

Reactor Coolant Pressure is used to determine whether SI should be terminated or reinstated and whether RCP operation should be continued. Knowledge of reactor coolant pressure is also used to maintain the proper relationship between RCS pressure and temperature and to establish correct conditions for RHR operations.

3,4. Reactor Coolant Outlet Temperature - T_{hot}, T_{cold} (Wide Range)

The reactor coolant wide range T_{hot} and T_{cold} channels are used to determine is SI should be terminated or reinstated and to establish core differential temperature during natural circulation. The wide range RCS temperature channels are also used to establish proper cooldown rates and correct conditions for RHR operation.

(continued)

BASES (continued)

LCOs (continued)

5. Reactor Vessel Water Level

The reactor vessel water level provides an indication of conditions inside the core during accident conditions.

6. Reactor Coolant System Subcooling Margin Monitor

The RCS subcooling margin monitor is used to determine the pressure and temperature margins to saturation of the primary coolant.

7. In Core Thermocouples

Core exit temperatures are used to verify adequate core subcooling to maintain the plant in a safe shutdown condition. A total of [16] thermocouples are provided in the core. Quadrants B and D each have [4] thermocouples. Quadrant A and C each have [4] thermocouples.

8, 9, 10. PORV, PORV Block Valve, and Safety Valve Position Indicator

To be provided.

11. Containment Pressure

Containment pressure is used to monitor margin to design pressure, monitor conditions inside containment following a break, and verify if accident is properly controlled.

12. Containment Sump Water Level

Containment sump water level is used to verify water source availability for the recirculation mode of ECCS operation after a LOCA. It may also be used to determine if a break is inside or outside containment or the potential for containment breach from very high water levels.

13. Refueling Water Storage Tank Level

RWST water level is used to verify the water source availability to the ECCS and Containment Spray Systems. It may also provide an indication of time for initiating cold leg recirculation following a LOCA.

(continued)

BASES (continued)

LCOs (continued)

14. Steam Generator Water Level (Narrow Range)

Steam Generator Water Level (Narrow Range) is used to verify that the steam generators are an adequate heat sink for the reactor. Narrow range steam generator level is also needed to make a determination on the nature of the accident in progress, e.g., verify a SGTR. Narrow range steam generator water level is used to verify plant conditions for termination of Safety Injection during secondary plant high energy line breaks outside containment.

15. Steam Generator Water Level (Wide Range)

Steam generator wide range water level is used to verify the existence of the steam generators as a heat sink. It is also used to monitor steam generator level trends when outside the narrow range span.

16. Steam Line Pressure

Steam line pressure is used to determine if a high energy secondary line rupture has occurred and the availability of the steam generators as a heat sink. It is also used to verify that a faulted steam generator is isolated. Steam line pressure may be used to ensure proper cooldown rates or to provide a diverse indication for natural circulation cooldown.

17. Auxiliary Feedwater Flow Rate

Auxiliary feedwater flow is used to determine if sufficient feedwater if being provided to maintain the heat sink and whether SI conditions may be terminated. It is also used to verify AFW isolation to a ruptured steam generator.

18. Essential Raw Cooling Water Flow

ERCW flow is used to determine the availability of the Ultimate Heat Sink.

(continued)

BASES (continued)

LCO (continued)

19. Containment Radiation Level

Containment radiation level is used to determine if a high energy line break has occurred, and whether the event is inside or outside of containment.

20. Shield Building Vent - High Range Noble Gas Monitor

The Shield Building Vent Radiation Monitor is used to monitor radioactivity release rates.

21. Steam Line Relief - Noble Gas Monitor

The steamline radiation monitors are used to detect primary to secondary leakage and monitor radioactivity release rates.

22. Condenser Vacuum Exhaust - High Range Noble Gas Monitor

The condenser vacuum exhaust radiation monitor is used to detect primary to secondary coolant leakage and provide indication of radioactivity release rates.

APPLICABILITY

Accident Monitoring Instrumentation LCO 3.3.3 is applicable in MODES 1, 2, and 3. The Accident Monitoring Instrumentation LCO provides the key variables to be monitored by the control room operators to perform the diagnosis and preplanned actions specified in the EOPs. These variables are related to the diagnosis and preplanned actions required to mitigate the design basis events, specifically, LOCA, Steam Line Break, Feedwater Line Break and Steam Generator Tube Rupture. These four events are assumed to occur in MODES 1, 2, or 3. The initial conditions of these events are based on the most limiting conditions of these MODES. Additionally, the accident monitoring instrumentation is qualified to perform its function under the worst case environs. Therefore, OPERABILITY of the accident monitoring instrumentation in MODES 1, 2, and 3 will ensure their operation from the initialization of the event through complete unit stabilization under the worst case environs. None of the design basis events are postulated to occur in MODES 4, 5, or 6, and therefore, the accident monitoring instrumentation is not required to be OPERABLE in these MODES.

(continued)

BASES (continued)

APPLICABILITY (continued)

Related LCOs

An accident monitoring instrumentation channel is composed of the field transmitter through the control room display. Portions of several accident monitoring instrumentation channels provide input to other systems for different functions. These related systems are;

- Reactor Trip System (RTS) Instrumentation, LCO 3.3.1,
- Engineered Safety Features Actuation System (ESFAS) Instrumentation, LCO 3.3.2, and
- Remote Shutdown System, LCO 3.3.4

ACTIONS

Category 1 accident monitoring instrumentation variables are required to meet single failure criteria in accordance with R.G. 1.97. Additionally, where failure of one accident monitoring channel results in information ambiguity, redundant displays disagree, additional information is provided to allow the operator to deduce the actual condition of the plant. This may be accomplished by providing diverse channel(s) or by providing additional independent channel(s) of the same variable (three channels versus two). If the number of inoperable channels for a particular function is greater than the number of inoperable channels addressed by the Condition statement, then the unit must then be placed in a MODE where the function is no longer required to be OPERABLE as per LCO 3.0.3.

Condition A

Condition A addresses the situation where one or more channels for one or more functions are inoperable at the same time. The Required Action is to refer to Table 3.3.3-1 and to take the Required Action(s) for the instrumentation function(s) affected. The Completion Time(s) are those from the referenced Condition(s) and Required Action(s).

Condition B

Condition B addresses the case where one of the Required Number of Channels for a particular 2 channel function is inoperable. The inoperable channel must be returned to an OPERABLE status within 7 days. The Completion Time of 7 days is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition C

Condition C addresses the case where two of the Required Number of Channels for a particular 2 channel function are inoperable. At least one channel must be restored to an OPERABLE status within 48 hours. The Completion Time of 48 hours is based on engineering judgment and industry operating experience. Restoration of one channel to an OPERABLE status will place the plant in Condition B.

Condition D

Condition D is similar to Condition C but applies to the RCS subcooling monitor. In addition to the Required Action of Condition C, the RCS subcooling must be monitored every 12 hours until the function is restored to OPERABLE status.

Condition E

Condition E addresses the OPERABILITY of the Incore Thermocouple (T/C) detectors. A minimum of [4] T/Cs per core quadrant are required OPERABLE for a total of [16] T/Cs. With less than 4 thermocouples per quadrant OPERABLE, the reliability of the [Inadequate Core Cooling Monitor (ICCM)] becomes questionable. A Completion Time of 48 hours is allowed to restore at least [4] T/Cs per core quadrant to an OPERABLE status. The Completion Time of 48 hours is based on engineering judgment and industry operating experience.

Condition F

Condition F addresses the functions that are on a per valve basis, PORV Position Indicator for example. The Required Number of Channels are 2/valve. Condition F addresses the case where one of the Required Number of Channels per valve is inoperable. The Required Action is to restore the channel to OPERABLE status within 7 days. Note that the Completion Time is on a per valve basis. For example, if a channel became inoperable on PORV A, 7 days would be allowed to restore that channel on that valve to an OPERABLE status. If another channel were to become inoperable on PORV B three days after the inoperable channel on PORV A, then 7 days would be allowed to restore the inoperable channel on PORV B. The 7 days allowed for PORV B is independent of the 7 days allowed for PORV A. The Completion Time of 7 days is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition G

Condition G addresses the functions that are on a per valve basis, PORV Position Indicator for example. The Required Number of Channels are 2/valve. Condition G addresses the case where two of the Required Number of Channels per valve are inoperable. The Required Action is to restore at least one channel to OPERABLE status within 48 hours. Note that the Completion Time is on a per valve basis. For example, if no channels were OPERABLE on PORV A, 48 hours would be allowed to restore at least one channel on that PORV to an OPERABLE status. If PORV B developed no channels OPERABLE one day after PORV A, then 48 hours would be allowed to restore at least one channel on PORV B to an OPERABLE status. The 48 hours allowed for PORV B is independent of the 48 hours allowed for PORV A. Restoration of one channel to an OPERABLE status will place the plant in Condition F. The Completion Time of 48 hours is based on engineering judgment and industry operating experience.

Condition H

Condition H addresses the functions that are on a per steam generator basis, Steam Generator Water Level for example. The Required Number of Channels is 1/steam generator. Condition H addresses the case where the Required Channel per steam generator is inoperable. The Required Action is to restore at least one channel to OPERABLE status within 48 hours. Note that the Completion Time is on a per steam generator basis. For example, if no channels were OPERABLE on Steam Generator A, 48 hours would be allowed to restore the channel on that steam generator to an OPERABLE status. If Steam Generator B developed no channels OPERABLE one day after Steam Generator A, then 48 hours would be allowed to restore the channel on Steam Generator B to an OPERABLE status. The 48 hours allowed for Steam Generator B is independent of the 48 hours allowed for Steam Generator A. The Completion Time of 48 hours is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition I

Condition I addresses the functions that are on a per steam generator basis, Steam Line Pressure for example. The Required Number of Channels are 2/steam generator. Condition I addresses the case where one of the Required Number of Channels per steam generator is inoperable. The Required Action is to restore the channel to OPERABLE status within 7 days. Note that the Completion Time is on a per steam generator basis. For example, if a channel became inoperable on Steam Generator A, 7 days would be allowed to restore that channel on that steam generator to an OPERABLE status. If another channel were to become inoperable on Steam Generator B three days after the inoperable channel on Steam Generator A, then 7 days would be allowed to restore the inoperable channel on Steam Generator B. The 7 days allowed for Steam Generator B is independent of the 7 days allowed for Steam Generator A. The Completion Time of 7 days is based on engineering judgment and industry operating experience.

Condition J

Condition J addresses the functions that are on a per steam generator basis, Steam Line Pressure for example. The Required Number of Channels are 2/steam generator. Condition J addresses the case where two of the Required Number of Channels per steam generator are inoperable. The Required Action is to restore at least one channel to OPERABLE status within 48 hours. Note that the Completion Time is on a per steam generator basis. For example, if no channels were OPERABLE on Steam Generator A, 48 hours would be allowed to restore at least one channel on that steam generator to an OPERABLE status. If Steam Generator B developed no channels OPERABLE one day after Steam Generator A, then 48 hours would be allowed to restore at least one channel on Steam Generator B to an OPERABLE status. The 48 hours allowed for Steam Generator B is independent of the 48 hours allowed for Steam Generator A. Restoration of one channel to an OPERABLE status will place the plant in Condition I. The Completion Time of 48 hours is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition K

Condition K addresses the Containment Radiation Level (High Range) instrumentation channels in accordance with NUREG-0737 and the NRC Staff Guidance on Technical Specifications for NUREG-0737 items. With less than two Channels OPERABLE, the Required Action is to establish an alternate method of Containment Radiation Level monitoring within 72 hours. The Completion Time of 72 hours to establish an alternate method of monitoring is considered a reasonable time. This is based upon the fact that the unit personnel will probably first assess the restoration feasibility and, if reasonable, attempt to restore the subject channels before establishing an alternate method of monitoring. Additionally, the unit will have procedures available to direct the personnel in establishing the alternate method of monitoring. These procedures will ensure the quick, orderly, establishment of, and logical utilization of, the alternate monitoring.

Condition L

Condition L addresses the Shield Building Vent, Condenser Vacuum Exhaust, and Steam Line Relief Noble Gas Monitors. With less than the Required Number of Channels OPERABLE, the Required Action is to establish an alternate method of monitoring within 72 hours. The Completion Time of 72 hours to establish an alternate method of monitoring is considered a reasonable time. This is based upon the fact that the unit personnel will probably first assess the restoration feasibility and, if reasonable, attempt to restore the subject channels before establishing an alternate method of monitoring. Additionally, the unit will have procedures available to direct the personnel in establishing the alternate method of monitoring. These procedures will ensure the quick, orderly, establishment of, and logical utilization of, the alternate monitoring.

(continued)

BASES (continued)

ACTIONS (continued)

Condition M

When the Required Actions are not met within the required Completion Time, a controlled shutdown is required by Required Action M.1. The unit must be placed in a MODE where the requirements of the LCO are no longer applicable. The 12 hours provided to be in MODE 4 is comprised of 6 hours to reach MODE 3 and 6 additional hours to reach MODE 4. The Completion Time of 6 hours to reach MODE 3 from MODE 1 is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action M.1, an additional 6 hours is a reasonable time, based on industry operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable, from MODE 3 without challenging unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

Surveillance Requirement 3.3.3.1 is the performance of a CHANNEL CHECK on each accident monitoring instrumentation channel every 31 days. A CHANNEL CHECK is simply the comparison of the indicated parameter values for each channel of a particular function. It is based upon the assumption that the two or three channels indicated in the control room should be reading approximately the same. Agreement is determined by the plant I&C and operations staff and should be based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the racks have drifted outside of their limit. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification. If the channels are normally off-scale during plant operations, then the CHANNEL CHECK will only verify that they are off-scale in the same direction. The surveillance Frequency of 31 days is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

Surveillance Requirement 3.3.3.2 is the performance of a CHANNEL CALIBRATION on each accident monitoring instrumentation channel every 18 months, or about every refueling. The 18 month interval is consistent with the assumptions of the magnitude of the transmitter drift used in the Setpoint Methodology Study. Additionally, the test is performed while the plant is shutdown because it may not be feasible to test all the transmitters while the plant is at power. The CHANNEL CALIBRATION is a complete check of the process control instrument loop and the transmitter. The transmitter "as found" value is noted and adjustments are made as necessary, with notation of the "as left" value for use in a drift calculation. The transmitter may be calibrated in place by use of deadweight or hydraulic testing equipment, on a bench using essentially the same type of equipment, or be replaced by an equivalent unit calibrated in a laboratory. Resistance temperature detectors may be calibrated in a test bath after removal from the piping, or replaced by a previously calibrated unit. The surveillance Frequency of 18 months is based on engineering judgment and industry operating experience.

REFERENCES

1. Watts Bar FSAR, Chapter [7].

2. U.S. NRC Regulatory Guide, RG-1.97, Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

3. U.S. NRC NUREG 0737, Clarification of TMI Action Plan Requirements

4. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants.

   a. GDC- 22, Protection System Independence

   b. GDC- 24, Separation of Protection and Control Systems

5. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The primary purpose of the remote shutdown instrumentation is to provide the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. A safe shutdown condition is defined as MODE 3. An examination of American Nuclear Society (ANS) Condition II, III, and IV events revealed that none require a cooldown to MODE 5 for safety related reasons (Ref. 1). The cooldown may be required to perform long term recovery, however, there is no safety reason to perform this cooldown in a limited amount of time. With the unit in MODE 3, the Auxiliary Feedwater System (AFS) and the steam generator safety valves or the steam generator atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFS and the ability to borate the Reactor Coolant System (RCS) from outside the Control Room allows extended operation in MODE 3. Additionally, nothing precludes the eventual achievement of MODE 5, even assuming a Safe Shutdown Earthquake (SSE), a loss of offsite power, and the most limiting active failure if arbitrary restrictions are not placed on either the time required to cool down or on the permissable operator actions outside the control room.

In the event that the control room becomes inaccessible, the operators can establish control in the Auxiliary Control Room, and place and maintain the unit in MODE 3. Not all controls and the necessary transfer switches are located at the Auxiliary Control Room Panels [ACRPs]. Some controls and transfer switches will have to operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a plant shutdown and can be maintained safely in MODE 3 for an extended period of time. The two [ACRPs] are electrically separated and are associated with the same safety-grade circuits that serve the respective trains. The design also provides for electrical separation of control and instrumentation circuits between the control room and [ACRP B]. Switches are provided on [ACRP B] to isolate and remove control from the control room for the Train [B] safe shutdown equipment necessary to place and maintain the plant in MODE 3 independent of the control room. This capability

(continued)

BASES (continued)

BACKGROUND (continued)

is assured in the event a fire causes damage in the control room and the subsequent evacuation of the operators. Train [B] controls and instrumentation were selected to be isolated because the controls and instrumentation for the turbine-driven auxiliary feedwater pump are on [ACRP B].

The remote shutdown instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

- Signal process control provides signal conditioning and a compatible electrical signal output to control board/control room/miscellaneous display devices.

- Display devices provide concise and accurate display of designated plant parameters for the period following an accident and/or seismic event.

The OPERABILITY of the remote shutdown instrumentation ensures that there is sufficient information available on selected plant parameters to place and maintain the unit in MODE 3 should the control room become inaccessible for any reason.

Field Transmitters and Instrumentation

In order to meet the design requirements for redundancy and reliability, more than one, and often as many as three, field sensors or transmitters are used to measure plant parameters. The Required Number of Channels for each plant parameter is listed in Table 3.3.4-1.

Transmitters are not usually tested while the unit is on line, and are normally subjected to maintenance only when recalibration is required during a unit shutdown, or when a failed instrument must be replaced. Prior to, and after installation, each field instrument is calibrated to provide the correct measured value and span. Surveillance requirements are discussed in the Surveillance Requirements section.

(continued)

BASES (continued)

BACKGROUND (continued)

Signal Process Control

The signal process control equipment provides signal conditioning and compatible output signals for instruments located on the main control board, in the main control room, or on the [ACRP]. The signal processing hardware is designed and configured to provide the required channel redundancy, channel separation, and channel independence.

Each channel of the process control equipment can be tested on-line to verify OPERABILITY. Surveillance requirements are discussed in the Surveillance Requirements section.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown Instrumentation LCO ensures the OPERABILITY of sufficient instrumentation and controls to place and maintain the plant in MODE 3 for an extended period of time from a location other than the Control Room in the event that the Control Room becomes inaccessible. The Remote Shutdown Instrumentation LCO does not satisfy any of the Selection Criteria of the NRC Interim Policy Statement (Ref. 2), but has been identified by the NRC as a risk significant item for retention in the Technical Specifications.

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from a location other than the Control Room. The instrumentation required is listed in Table 3.3.4-1. The table contains the types of variable commonly found and includes variables that utilize each of the prescribed Required Actions. Plant-specific lists will be developed when the specification is made plant-specific. The controls and transfer switches can be found in [FSAR] Chapter [7] (Ref. 1). The controls and transfer switches are those required for safety grade:

- Core reactivity control,
- Reactor Coolant System pressure control,
- Decay heat removal via the Auxiliary Feedwater System and the steam generator safety valves or steam generator atmospheric dump valves,

(continued)

BASES (continued)

LCO (continued)

- Decay heat removal via the Residual Heat Removal System,
- Reactor Coolant System inventory control via charging flow, and
- Safety support systems for the above functions.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room. With the unit in MODES 4, 5, or 6, ther is no requirement to place and maintain the unit in MODE 3. Therefore, this LCO is not applicable in MODES 4, 5, or 6.

ACTIONS

The Conditions and associated Required Actions maintain the required number of instrumentation channels, controls, and transfer switches to place and maintain the unit in MODE 3 for an extended period of time from a location other than the control room. If the number of inoperable channels for a particular function is greater than the number of inoperable channels addressed by the Condition statement, then the plant must be placed in a MODE where the LCO is no longer applicable as per LCO 3.0.3.

Condition A

Condition A addresses the situation where one or more channels for one or more functions are inoperable at the same time. The Required Action is to refer to Table 3.3.4-1 and to take the Required Action(s) for the instrumentation function(s) affected. The Completion Time(s) are those from the referenced Condition(s) and Required Action(s).

Condition B

Condition B addresses the case where one or more channels for a particular function are inoperable. The inoperable channel(s) must be returned to an OPERABLE status within 30 days. The Completion Time of 30 days is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

ACTIONS (continued)

Condition C

Condition C addresses the case where one or more channels per loop for a particular function are inoperable. The inoperable channel must be returned to an OPERABLE status within 30 days. The Completion Time of 30 days is based on industry operating experience.

Condition D

Condition D addresses the reactor trip breaker position indication circuits. If one or more position indication circuit(s) are inoperable, 30 days is allowed to restore the circuit(s) to an OPERABLE status. Note that the Completion Time is on a per reactor trip breaker basis. For example, if the indication circuit for one of the reactor trip breakers becomes inoperable, 30 days is allowed to restore it to an OPERABLE status. If the indication circuit for the other reactor trip breaker becomes inoperable three days after the first one, then 30 days is allowed to restore the second indication circuit to an OPERABLE status. The 30 days allowed for the first indication circuit is independent of the 30 days allowed for the second indication circuit. The Completion Time of 30 days is based on engineering judgment and industry operating experience.

Condition E

Condition E addresses the case where one or more channels for rod bottom bistables are inoperable. The inoperable channel(s) must be returned to an OPERABLE status within 30 days. The Completion Time of 30 days is based on engineering judgment and industry operating experience.

Condition F

Condition F addresses the functions that are on a per steam generator basis, Steam Generator Water Level for example. Condition F addresses the case where one or more channel(s) per steam generator are inoperable. The Required Action is to restore the channel(s) to OPERABLE status within 30 days. Note that the Completion Time is on a per steam generator basis. For example, if a channel became inoperable on Steam Generator A, 30 days would be allowed to

(continued)

BASES (continued)

ACTIONS (continued)

Condition F (continued)

restore that channel on that steam generator to an OPERABLE status. If another channel were to become inoperable on Steam Generator B three days after the inoperable channel on Steam Generator A, then 30 days would be allowed to restore the inoperable channel on Steam Generator B. The 30 days allowed for Steam Generator B is independent of the 30 days allowed for Steam Generator A. The Completion Time of 30 days is based on engineering judgment and industry operating experience.

Condition G

Condition G addresses the case where one or more channels per train for a particular function are inoperable. The inoperable channel(s) must be returned to an OPERABLE status within 30 days. The Completion Time of 30 days is based on engineering judgment and industry operating experience.

Condition H

Condition H addresses the inoperability of one or more transfer switches or control circuits. The Required Action is to restore the transfer switches and control circuits to an OPERABLE status within 30 days. Note that the Completion

Time is on a per transfer switch or per control circuit basis. For example, if transfer switch A became inoperable, 30 days would be allowed to restore it to an OPERABLE status. If control circuit C became inoperable 4 days after transfer switch A, then 30 days would be allowed to restore control circuit C to an OPERABLE status. The 30 days allowed for control circuit C is independent of the 30 days allowed for transfer switch A. The Completion Time of 30 days is based on engineering judgment and extensive plant operating experience.

Condition I

When Conditions B, C, D, E, F, G, or H are not met within the
required Completion Time, a controlled shutdown is required by Action I.1. The unit must be placed in a MODE where the requirements of the LCO are no longer applicable. The 12 hours provided to be in MODE 4 is comprised of 6 hours to reach MODE 3 and 6 additional hours to reach MODE 4. The Completion Time of 6 hours for reaching MODE 3 from MODE 1

(continued)

BASES (continued)

ACTIONS (continued)

Condition I (continued)

is reasonable based on industry operating experience and normal cooldown rates, and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action I.1, an additional 6 hours is a reasonable time, based on industry operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable, from MODE 3 without challenging unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

Surveillance Requirement 3.3.4.1 is the performance of a CHANNEL CHECK on each remote shutdown instrumentation channel every 31 days. A CHANNEL CHECK is simply the comparison of the indicated parameter values for each channel of a particular function. It is based upon the assumption that the two or three channels indicated in the control room should be reading approximately the same. Agreement is determined by the plant I&C and operations staff and should be based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the racks have drifted outside of their limit. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification. If the channels are normally off-scale during plant operations, then the CHANNEL CHECK will only verify that they are off-scale in the same direction. The surveillance Frequency of once per 31 days is based on engineering judgment and industry operating experience.

SR 3.3.4.2

Surveillance Requirement 3.3.4.2 verifies that each Remote Shutdown System transfer switch and control circuit performs its intended function. This verification is performed from the [ACRPs] and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the [ACRPs] and the local control stations. The surveillance Frequency of every 18 months is based on engineering judgment and industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.3

Surveillance Requirement 3.3.4.3 is the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 18 months, or approximately every refueling. The 18 month assumption is made in the determination of the magnitude of the transmitter drift in statistical analyses. The test is a complete check of the process control instrument loop and the transmitter. The transmitter "as found" value is noted and adjustments are made as necessary, with notation of the "as left" value for use in a drift calculation. A completion of this test results in the channel being properly adjusted and expected to remain within the allowable tolerance until the next scheduled surveillance.

REFERENCES

1. Watts Bar FSAR, Chapter [7]

2. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications, May 9, 1988."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature and Flow DNB Limits

BASES

BACKGROUND

This bases addresses requirements for maintaining RCS pressure, temperature, and flow within the limits assumed in the safety analysis for the Limiting Conditions for Operation (LCOs) and Surveillance Requirements of this LCO.

The safety analyses (Ref. 1) assume initial conditions within the normal steady state envelope of operating conditions. The limits placed on RCS pressure, temperature, and flow ensure that the minimum Departure from Nucleate Boiling Ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit 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 pressure lower than the minimum specified will cause the unit to approach DNBR limits.

The RCS coolant average temperature limit specified is consistent with full power operation within the nominal operational envelope. Indications of T_{avg} are averaged to come up with a value for comparison to the limit. A higher average temperature will cause the unit to approach DNBR limits.

Westinghouse unit operational experience has demonstrated that RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit specified 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 value of RCS flow will cause the unit to approach DNBR limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in the event that a DNB-limited event were to occur.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB-limited transients analyzed in the unit safety analyses (Ref. 1).

The safety analyses have shown that transients initiated from the limits of LCO 3.4.1 will result in meeting the DNBR criteria.

The DNB parameters are process variables which are initial conditions of design basis accidents or transient analyses that either assume the failure of or present a challenge to the integrity of a fission product barrier. As such, they satisfy the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

LCO 3.4.1 provides limits on the monitored process variables pressurizer pressure, RCS average temperature, and RCS flow rate to ensure that the core operates within the limits assumed in the unit safety analysis. Operating within these limits will result in meeting DNBR criteria in the event of a DNB-limited transient.

APPLICABILITY

The limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state pump operation in MODE 1 in order to assure that DNBR criteria will be met in the event of a loss of coolant flow or other DNB-limited transient. The DNBR criteria for DNB-limited transients initiated from lower MODES are satisfied by meeting the requirements of LCOs for the pressurizer, the Reactor Trip System, the Engineered Safety Features Actuation System, and the RCS loops.

The limit on RCS pressure is not applicable during short-term operational transients such as a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER (RTP) per minute or a THERMAL POWER step increase of greater than 10% of RTP.

(continued)

BASES (continued)

APPLICABILITY (continued)

These conditions represent short-term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels less than RTP, there exists an increased DNBR margin to offset the temporary pressure variations.

Another set of limits on DNB-related parameters is provided in SL 2.1, Reactor Core Safety Limits. Those limits are less restrictive than the limits of LCO 3.4.1, but violation of safety limits merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, SL 2.1 is provided as a Cross-Reference because the operator should check whether or not a Safety Limit may have been exceeded.

Both LCO 3.4.1 and LCO 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor, have a common objective - keeping the DNBR above its minimum value. Because of this commonality, LCO 3.2.2 is included in the Cross-Reference section.

ACTIONS

A.1

RCS pressure and core average coolant temperature are controllable and measurable parameters. If these parameters cannot be controlled to within the LCO limits, then power must be reduced to restore the margin of safety and eliminate the potential for violation of the accident analysis bounds.

RCS Flow cannot be changed but it is indicated to determine if it is within the limits. If the indicated flow rate is below the LCO limit, a short time is allowed for determining whether this indication is due to an instrument error. If this is the case, it may be possible to ascertain what the flow is by other means, such as the unit computer. If this is not the case, then power must be reduced to eliminate the potential for violation of the accident analysis bounds.

Required Actions A.1 above all have a Completion Time of 2 hours which is intended to provide sufficient time to determine the cause of the violation of a LCO limit and to correct the error. In addition, the probability of a DNB-limited event during the completion time interval is acceptably low.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If any of the DNB parameters exceed their limits and cannot be restored within limits in 2 hours, the unit must be placed in a MODE where the limits no longer apply. This is done by placing the unit in MODE 2 within 6 hours. The Completion Time of 6 hours to reach MODE 2 is a reasonable time, based on industry operating experience, to reach MODE 2 from full power without challenging safety systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1, 3.4.1.2, and 3.4.1.3

The surveillance every 12 hours of DNB-related parameters through instrument readout is sufficient to ensure that the parameters are maintained within their limits following load changes and other expected transient operations.

Typically renormalized after each refueling upon reaching 75% to 100% of RTP, these flow meters still give a reasonable indication of flow, providing assurance of not exceeding any DNB related flow limits.

SR 3.4.1.4

Performance of a precision heat balance once every 18 months allows the RCS flow to be determined from measurements and verifies that the actual RCS flow is $\geq$ the minimum required RCS design flow. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation. This frequency has been shown to be acceptable through operating experience.

REFERENCES

1. Watts Bar FSAR, Section 6.2 and 15.2.
2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Minimum Temperature For Criticality

BASES

BACKGROUND

The minimum temperature for reactor criticality is based upon several considerations. These include ensuring that the reactor vessel is above its minimum Reference Temperature for Nil-Ductility Temperature (RT_{NDT}) when the reactor is critical and that the temperature will be within the narrow range instrumentation spans. In addition, the minimum temperature for criticality provides constraints on the Moderator Temperature Coefficient (MTC) although the MTC LCO may be more restrictive.

APPLICABLE SAFETY ANALYSES

All low power safety analyses assume initial RCS loop temperatures ≥ HZP, [557]°F (Ref. 1). The temperature for HZP is a process variable that is an initial condition of Design Basis Accidents (DBA) such as the Rod Cluster Control Assembly (RCCA) withdrawal, RCCA ejection and main steam line break accidents performed at zero power that either assume the failure of or present a challenge to the integrity of a fission product barrier. The minimum temperature for criticality limitation provides a small band [6]°F for critical operation below hot zero power (HZP). This band allows critical operation below HZP during unit 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.

Although the RCS minimum temperature for criticality is not a process variable that is an initial condition assumed in a DBA, it is a process variable monitored by the operators during startup, and is displayed on permanently installed instrumentation in the control room. As such, the RCS minimum temperature for criticality satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

(continued)

BASES (continued)

LCOs

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

The reactor can only be critical ($k_{eff} \geq 1.0$) in MODES 1 and 2. Therefore, the SR is applicable to MODE 1 and to MODE 2 (with $k_{eff} \geq 1.0$); and further, only when any RCS average loop temperature is < [56]°F and the T_{avg} - T_{ref} deviation alarm is in an alarm state. In the range of [551]°F (the minimum temperature for criticality) to [554]°F there is a potential for RCS loop average temperature to fall below the LCO requirement. Below [557]°F, T_{ref} is essentially constant and equal to [557]°F ($T_{no\ load}$). Therefore, a T_{avg} - T_{ref} deviation alarm would be due to movement of RCS loop average temperature below $T_{no\ load}$ and would alarm from [0 to 3]°F above the minimum temperature for criticality. As power level increases RCS loop average temperatures increase to a value far enough above [561]°F, that the potential for RCS loop average temperatures to fall below [551]°F is so diminished that the LCO is no longer applicable.

The special test exception in the Cross-References permits PHYSICS TESTS to be performed at less than or equal to 5.0% of RATED THERMAL POWER 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 requirement of this LCO.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

When any RCS loop average temperature is < [551]°F, the actions are to restore all RCS loop average temperatures to within the specified limit within 15 minutes or place the unit in MODE 3 within the next 15 minutes. The LCO is no longer applicable in MODE 3. The Completion Time of 15 minutes is based on engineering judgment and the practical amount of time that it may take to perform the Required Action. Operation with the reactor critical and below [551]°F could violate the assumptions for the steam line break accident and other accidents analyzed in the safety analyses. The additional 15 minutes to place the unit in MODE 3 is based on unit operating experience, and should not place unnecessary stress on unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

Verification that all RCS loop average temperatures are ≥ [551]°F will provide assurance that the initial conditions of the safety analyses are not violated. The requirement to verify that all RCS loop average temperatures are above [551]°F 15 minutes before the expected time to achieve criticality allows the operator time to adjust the temperatures or delay criticality so the LCO will not be violated. When any RCS loop average temperature is less than [561]°F and the T_{avg} - T_{ref} deviation alarm is not reset, RCS loop average temperatures could fall below the LCO requirement without additional warning. The Surveillance Requirement to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.

REFERENCES

1. Watts Bar FSAR, Chapter 15, Accident Analysis.

2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure/Temperature (P/T) Limits

BASES

BACKGROUND

Four P/T limit curves for heatup, cooldown, criticality, and Inservice Leak and Hydrostatic Test (ILHT), and data for the maximum allowable rate-of-change of reactor coolant temperature are shown in the RCS PRESSURE/TEMPERATURE LIMITS REPORT. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. These curves and this specification establish operating limits that provide a margin to brittle failure of major components of the Reactor Coolant Pressure Boundary (RCPB). The reactor vessel, outlet nozzle, and head are the components for which the technical specification limits are most pertinent.

The origin of these pressure and temperature limits is found in Appendix G to 10 CFR Part 50 (Ref. 1). Appendix G requires that limits be established and the limits shall be based on specific fracture toughness requirements for RCPB materials such that an adequate margin to brittle failure will be provided during normal operation, system hydrostatic tests, and anticipated operational occurrences. Reference 1 mandates the use of ASME Section III, Appendix G (Ref. 2).

This specification provides two types of limits:

a. RCS maximum pressure and minimum temperature curves that define allowable operating regions, and

b. Limits on the allowable RCS temperature rate-of-change which limits thermal gradients through the vessel wall and thus limits tensile stresses in the vessel wall.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses for those portions of the reactor vessel and head that are most restrictive. At any specific pressure, temperature, and temperature rate-of-change, one location within the geometry of the reactor vessel or head will dictate the most restrictive limit. Across the entire span of the P/T limit curves, different locations are most restrictive and thus the curves are a composite of the most restrictive regions.

(continued)

BASES (continued)

BACKGROUND (continued)

The heatup curves represent a different set of restrictive elements than the cooldown curves because the direction of the thermal gradients through the vessel wall is reversed. The thermal gradient reversal tends to alter the location of the tensile stress between the outer and inner walls.

The ILHT curve values are based on different calculation safety factors (Ref. 2) than the heatup and cooldown curves. The ILHT curves also extend to the RCS design pressure [2500] psia to bound the test range.

[The criticality limit curve is based on Reference 1 requirements that the curve be no less than 40°F above the heatup or cooldown curves, nor lower than the minimum permissible temperature for the ILHT. However, this criticality limit curve is not operationally limiting, because a more restrictive limit on Minimum Temperature for Criticality exists in LCO 3.4.2, RCS Minimum Temperature For Criticality.]

The P/T limit curves and associated temperature rates-of-change have been developed in conjunction with stress analyses, and provide a conservative margin to non-ductile failure. The consequence of violating the LCO limits is that the RCS has operated under conditions which could have resulted in brittle failure of the RCPB, possibly leading to an non-isolable leak or loss of coolant accident. In the event that these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the reactor vessel. ASME Section XI Appendix E (Ref. 3) provides a recommended methodology for evaluating operating events which cause an excursion outside the normal limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from design bases accident events presented in the FSAR, but are prescribed as guidance used during normal operation to avoid encountering pressure, temperature and temperature rate-of-change conditions which might cause undetected flaws to propagate resulting in non-ductile failure of the RCPB. Linear elastic fracture mechanics methodology, following the guidance given by 10 CFR Part 50 Appendix G (Ref. 1), ASME Section III Appendix G, (Ref. 2) and Regulatory Guide 1.99, Rev. 2 (Ref. 4), is used to determine the stresses and the material RT_{NDT} at locations within the RCPB.

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BASES (continued)

APPLICABLE SAFETY ANALYSIS (continued)

Although any region within the pressure boundary is subject to non-ductile failure, the regions that provide the most restrictive limits are the reactor vessel closure head, the reactor outlet nozzles, and the reactor vessel beltline. With increasing neutron fluence the reactor vessel beltline typically becomes the most restrictive region.

Fracture toughness properties of the ferritic materials of the reactor vessel are determined in accordance with the NRC Standard Review Plan (Ref. 5), ASTM E185 (Ref. 6), and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 2.

The concern addressed by Reference 2 is that undetected flaws could exist in the RCPB components, which if subjected to unusual pressure and/or thermal stresses could result in non-ductile failure. Unusual RCS P/T combinations can cause stress concentrations at flaw locations which can tend to cause flaw growth resulting in failure before the ultimate strength of the material is attained. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness (T), 1/4 T, and a length of 3/2 T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack are well within the current detection capabilities of inservice inspection techniques. Therefore, the P/T limit curves developed for this postulated crack are conservative and provide sufficient safety margins for protection against non-ductile failure.

Flaw growth is resisted by the material toughness, which is a property that varies with temperature and is lower at room temperature than at power operation. Furthermore, the material toughness is affected by neutron fluence and the chemistry of the base metal and welds. Neutron fluence causes the steel toughness to decrease and is cumulative, with the effect that toughness steadily decreases with exposure time. Regulatory Guide 1.99, Rev. 2 provides guidance for evaluating the effect of neutron fluence.

An indicator of the temperature effect on toughness is the Nil-Ductility Temperature, NDT. The NDT for the steel alloy used in vessel fabrication has been established by

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

testing. The NDT is a temperature below which brittle failure may occur and above the NDT ductile failure may occur. However, the exact temperature value can not be established with precision. Consequently a Nil-Ductility Reference Temperature, RT_{NDT}, has been established for use.

To assure that the radiation embrittlement effects on the RT_{NDT} are accounted for in the calculation of the limit curves, the most limiting RT_{NDT} (of the various reactor vessel components) is used and includes a radiation induced shift corresponding to the end of the period for which heatup and cooldown curves are generated.

This shift is a function of both neutron fluence and the copper and nickel content of the vessel material. The heatup and cooldown limit curves include predicted adjustments for shift in RT_{NDT}, and state the number of effective full power years for which this shift applies.

Values for shift in RT_{NDT} are updated periodically after the results from the material surveillance program are available. Surveillance capsules will be removed and evaluated in accordance with the requirements of ASTM E185 and 10 CFR Part 50, Appendix H (Ref. 7). The heatup and cooldown curves must be recalculated before the shift in RT_{NDT} determined from the surveillance capsule exceeds the calculated RT_{NDT} shift for the equivalent capsule radiation exposure.

With the material toughness established as a function of RT_{NDT}, stress analyses are performed per Reference 2 to set the pressure and temperature limits. The limiting location of maximum stress may vary during both heatup and cooldown operations, depending on pressure, temperature, and temperature rates-of-change. The P/T limit curves must also account for a requirement from Reference 1 which states that the minimum metal temperature of the closure head flange and vessel flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure. Thus, the heatup and cooldown curves are composites of the limiting pressures at specific temperatures, with separate curves derived for varying heatup and cooldown rates. [The limit curves are also adjusted for instrumentation errors associated with the wide range pressure and temperature instruments.]

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The P/T limits are a process variable that is an initial condition of a design basis accident or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, they satisfy the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 8).

LCOs

The two elements of the LCO are:

a. The limit curves for heatup, cooldown, criticality, and ILHT.

b. Limits on the temperature rate-of-change.

The P/T limit curves define allowable operating regions which provide a margin to non-ductile failure. The limits on rate-of-change of temperature control the thermal gradient through the walls and are used as input for calculating the heatup, cooldown and ILHT limit curves. Thus operations within the temperature rate-of-change limits ensures the validity of the P/T limit curves.

APPLICABILITY

The potential for violating the P/T or temperature rate-of-change limits exists at all times. Thus this LCO is applicable during all MODES. However, during MODES 1 and 2, LCO 3.4.2, RCS Minimum Temperature For Criticality, prescribes a minimum RCS temperature for criticality and will be more limiting than this LCO. Other cross references include: SL 2.1, Reactor Core Safety Limits, which sets the limits on THERMAL POWER, pressurizer pressure, and T_{avg} during MODES 1 and 2; SL 2.2, Reactor Coolant System Pressure Safety Limit, which sets a maximum limit on RCS pressure during MODES 1 through 5; and LCO 3.4.1, RCS Pressure, Temperature and Flow DNB Limits, which sets limits on RCS pressure, T_{avg}, and flow during MODE 1.

Also, overpressure prevention is available to prevent violation of the P/T limits during all MODES. This overpressure prevention is provided by the pressurizer safety valves at RCS temperatures above [310]°F, and by a Cold Overpressure Prevention System at lower temperatures. LCO 3.4.16, Cold Overpressure Mitigation System, provides the OPERABILITY requirements for the Cold Overpressure Mitigation System.

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BASES (continued)

ACTIONS

A.1 and A.2

Condition A addresses operation outside the acceptable region of the P/T limit curves and exceeding heatup and cooldown rates. Increasing temperature or reducing pressure per Required Action A.1 are direct ways to restore the RCS to an acceptable operating region so the combined temperature and pressure stresses are reduced. While increasing temperature or decreasing pressure, the heatup or cooldown rate limit should not be exceeded.
The Completion Time of 30 minutes is based on engineering judgment as to a reasonable time to restore the proper pressure and temperature. Most violations will not be so severe that the activity cannot be accomplished in this time and in a controlled manner.

In addition to increasing temperature or reducing pressure, an evaluation to determine if RCS operation may proceed is required by Required Action A.2. This evaluation has to be performed independent of whether the unit may have returned to within the limits of the LCO. ASME Section XI Appendix E (Ref. 4) may be used to support this evaluation. The Completion Time of 6 hours for this evaluation is based on the Completion Time allowed to reach MODE 3 per Required Action B.1, as discussed below. For a mild violation the evaluation should be possible in this time. More severe violations may require special, event specific stress analyses, and/or inspections which are appropriately carried out while the RCS is in a reduced pressure and temperature condition.

B.1 and B.2

Required Actions B.1 and B.2 address the failure to perform Required Actions A.1 or A.2 within the allowed Completion Times. Reducing the operational MODE is considered a prudent action because: a) the RCS remained in an unacceptable region for an extended period of increased stress, or b) 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 while the RCS is in a low pressure and temperature state. With the unit at reduced pressure conditions, the possibility of propagation of undetected flaws is reduced.

The Completion Time of 6 hours to reach MODE 3 is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is an appropriate surveillance when RCS temperature and pressure conditions are undergoing planned changes. The frequency of 30 minutes is based on engineering judgment. Since temperature rate-of-change limits are specified in hourly increments, a half hour time period permits assessment and correction for minor deviations within a reasonable time.

There are no Surveillance Requirements stated for the P/T limits during criticality because LCO 3.4.2, RCS Minimum Temperature For Criticality, contains a more restrictive LCO, with its own Surveillance Requirements.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements.", 1986.

2. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

3. American Society of Mechanical Engineer (ASME), Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events", 1986.

4. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," United States Nuclear Regulatory Commission, May 1988.

5. NUREG-0800, USNRC Standard Review Plan, Section 5.3.1, "Reactor Vessel Materials", Rev. 1, July 1981.

6. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

7. Title 10 Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements.", 1986.

8. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - Modes 1 and 2

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, 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,

d. providing a second barrier against fission product release to the environment, and

e. removal of the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing a steam generator, a reactor coolant pump, and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The steam generators provide the heat sink to the isolated secondary coolant. The reactor coolant pumps circulate the coolant through the reactor vessel and steam generators at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures boration and chemistry control.

APPLICABLE SAFETY ANALYSES

The unit is designed to operate with all reactor coolant loops in operation to maintain Departure from Nucleate Boiling Ratios (DNBR) above the safety limit or limits as applicable during all normal operations and anticipated transients. By assuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	the fuel cladding and the reactor coolant, thus preventing fuel cladding damage. The RCS is part of the primary success path and which functions or actuates to 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. As such, it satisfies Criterion 3 of the NRC Interim Policy Statement (Ref. 1).
LCOs	A loss of forced circulation could cause a Departure from Nucleate Boiling (DNB) and the potential for fuel damage. Therefore, this LCO requires that all RCS loops be OPERABLE and in operation in MODES 1 and 2, to ensure adequate heat transfer, heat sink, and boration control of the reactor coolant.
APPLICABILITY	In MODES 1 and 2 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, all RCS loops are required to be in operation in these MODES to prevent DNBR core damage concerns. Maximum decay heat production is approximately [8]% of RATED THERMAL POWER. As such, the forced circulation flow and heat sink requirements are reduced for lower, non-critical MODES as indicated by the LCOs of MODES 3, 4 and 5.
ACTIONS	<u>A.1</u> A partial loss of coolant flow with the reactor at power will cause a rapid increase in the coolant temperature. This increase could result in DNB concerns with subsequent fuel damage. For this reason it is necessary that, when the reactor is in MODES 1 and 2, all the heat transfer and heat sink capability from the core be available, i.e., all the loops be in operation. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verifying RCS loop operation. The verification may be performed by checking RCS pump operation and RCS flow and temperature monitoring instrumentation. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through operating experience.

SR 3.4.4.2

This SR requires the steam generator to be OPERABLE in accordance with the Inservice Inspection and Testing Program and the Steam Generator Tube Inspection Program.

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - Mode 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generator, to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid and fission product barrier.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing a steam generator, a reactor coolant pump, and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The steam generators provide the heat sink. The reactor coolant pumps circulate the water through the reactor vessel and steam generators at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers are in the closed position, the possibility of an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the Rod Control System. In addition, with the reactor trip breakers in the closed position, thus enabling some rods to be partially or totally withdrawn from the core, the possibility of a power excursion due to the ejection of an inserted control rod is possible. Such a transient could be caused by the mechanical failure of a control rod drive mechanism housing.

Therefore, in MODE 3 with reactor trip breakers in the closed position, accidental control rod withdrawal from subcritical or rod ejection events are postulated, and require at least [two] RCS loops in operation to ensure that the safety analyses limits are met.

The RCS is part of the primary success path and which functions or actuates to 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. As such, it satisfies Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

(continued)

BASES (continued)

LCOs

The LCO stipulates that at least [two] of the RCS loops be OPERABLE. In MODE 3 with the reactor trip breakers in the closed position, [two] RCS loops must be in operation. [Two] RCS loops are required to be in operation in MODE 3 with reactor trip breakers closed due to the postulation of a power excursion because of an inadvertent control rod withdrawal or rod ejection. [Two] RCS loops in operation ensure that the safety limit criteria will be met.

With the reactor trip breakers in the open position, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and ensure homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to meet single failure considerations.

The note in the LCO statement provides for a limited amount of natural circulation heat removal. The allowable duration is 1 hour, under the following conditions:

a. No operations which would cause dilution of the RCS boron concentration, therefore maintaining the margin to criticality.

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.

APPLICABILITY

The most stringent condition of the LCO, [two] RCS loops OPERABLE and [two] reactor coolant loops in operation, applies to MODE 3 with reactor trip breakers in the closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the reactor trip breakers open. The required heat sinks for MODES 1 and 2, and MODES 4 and 5 are covered by LCO 3.4.4 through LCO 3.4.8.

(continued)

BASES (continued)

ACTIONS

A.1

If only one RCS loop is OPERABLE, there may not be enough reserve heat removal capacity with forced flow to ensure that there will not be a safety limit concern with the reactor trip breakers in the closed position for a postulated rod withdrawal from subcriticality or rod ejection event.

The Required Action is restoration of the RCS loops to OPERABLE status within a Completion Time of 72 hours. This time allowance is based on engineering judgment and the consideration that a single loop has a heat transfer capability considerably greater than needed to remove the decay heat produced in the reactor core.

A.2

If restoration to the status of at least [two] RCS loops OPERABLE is not possible within 72 hours, the unit must be placed in MODE 4. In MODE 4 the heat removal requirements are reduced and the unit 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.

B.1 and B.2

When the reactor trip breakers are in the closed position, it is postulated that a power excursion may occur as the result of an inadvertent control rod withdrawal or rod ejection event. This mandates having the heat transfer capacity of [two] RCS loops in operation. If only one loop is in operation, the reactor trip breakers must be opened. The Completion Time of 1 hour for opening the breakers is adequate to perform this operation in an orderly manner.

(continued)

BASES (continued)

ACTIONS (continued)

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

If no loop is in operation, except as provided in Note 1 in the LCO section, the reactor trip breakers must be opened immediately and all operations involving a reduction of RCS boron concentration must be immediately suspended. This is necessary because boron dilution requires forced circulation for proper homogenization and opening the reactor trip breakers removes the possibility of an inadvertent rod withdrawal from subcritical. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

Steam generator OPERABILITY is verified by ensuring that the secondary-side water level is ≥ [10]% on the wide range indicators. If this level is not maintained, the primary tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The frequency of 12 hours is based on engineering judgment on the wide range indicators. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.5.2

RCS loop operation verification is performed by proper circuit alignment, RCS flow rate and temperature monitoring. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.5.3

Verification that the required number of reactor coolant pumps are OPERABLE ensures that single failure criteria is met. The requirement also ensures that additional reactor coolant pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. The frequency of 7 days ensures that the required flow can be made available and is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.5.4

Refer to Bases for SR 3.4.4.4 (Steam Generator OPERABILITY).

(continued)

BASES (continued)

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

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) System 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 [four] RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a Reactor Coolant Pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The RCPs circulate the water through the reactor and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, the function of the reactor coolant is to transport decay heat from the reactor core either to the SGs or the RHR heat exchangers, where heat exchange with the secondary systems takes place. This is achieved in a forced circulation mode.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

The RCS and RHR System are part of a primary success path and which function or actuate to 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. As such, they satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

(continued)

BASES (continued)

LCOs

The LCO requires that at least two of the RCS and/or RHR loops be OPERABLE in MODE 4. Any one loop in operation provides enough flow capacity to remove the decay heat from the core with forced circulation. The second loop which is required to be OPERABLE meets single failure criteria.

Note 1 in the LCO statement provides for a limited amount of natural circulation heat removal. The period of time allowed is 1 hour because operating experience has shown that boric acid stratification is not a problem when natural circulation is limited to 1 hour. Part 2 of Note 1 requires core outlet temperature to be maintained at least 10°F below saturation temperature and addresses the concern that a bubble might form and restrict flow in the RCS under natural circulation conditions.

Note 2 in the LCO statement restricts the start of a RCP under specified conditions. This restraint is included to prevent a cold overpressure event due to a thermal transient.

APPLICABILITY

Forced circulation is provided for the reactor coolant to remove decay heat from the core and to provide proper boron homogenization in MODE 4. One loop of either RCS or RHR provides sufficient circulation for these purposes. This condition is required until the unit is placed in MODE 5. In MODE 5 the decay heat removal is provided only by RHR because it is not feasible to use the RCS since the RCS coolant temperatures are below the secondary-side saturation point.

ACTIONS

A.1

If only one RHR loop is OPERABLE and no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to satisfy single failure considerations. The Completion Time of 1 hour is based on engineering judgment and the fact that natural circulation cooldown is allowed for 1 hour. If a second loop cannot be made OPERABLE within the required Completion Time, the unit must be placed in a MODE in which these requirements do not apply. This is accomplished by placing the unit in MODE 5 within an additional 24 hours.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

If only one loop RCS is OPERABLE and no RHR loop is OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to satisfy single failure considerations.

C.1 and C.2

If no loop is in operation, except as provided in Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be immediately suspended, because boron dilution requires forced circulation for proper homogenization, and the margin to criticality must not be reduced in this type of operation. In addition, action must immediately be initiated to restore one loop to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

If the SG water level is < [10]% (wide range indication), the primary tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

Verifying that at least one RCP or RHR loop is in operation ensures that the reactor coolant is in a forced circulation mode and capable of transferring decay heat to either the associated SG or RHR heat exchanger. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.6.3

Verifying that the required RCPs and/or RHR pump(s) are OPERABLE ensures that single failure criteria is met. The requirement also ensures that additional RCPs and/or RHR pump(s) can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. The frequency of 7 days ensures that the required flow can be made available and is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.6.4

This SR refers back to SR 3.4.4.2 which requires that the steam generator be OPERABLE in accordance with the Steam Generator Tube Inspection Program (5.9.11) and the Inservice Inspection and Testing Program (5.9.14).

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

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 functions of the reactor coolant are: 1) to remove decay heat generated in the fuel, and to transfer this heat to the component cooling water via the Residual Heat Removal (RHR) heat exchangers; and 2) to prevent stratification of the soluble neutron poison, boric acid. The reactor coolant is circulated by means of [two] RHR loops connected to the reactor vessel, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for both control and protection. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

If only one RHR loop is OPERABLE and in operation, [two] Steam Generators (SGs) with levels above [10]% (wide range indication) provide an alternate method for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

The RHR System is part of a primary success path and which functions or actuates to 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. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

In MODE 5 with the RCS loops filled, the LCO requires that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or [two] SGs with water level $\geq$ [10]%. One RHR loop injecting into two cold legs provides sufficient forced circulation to perform the safety functions of the reactor coolant under these unit conditions. The second RHR loop

(continued)

BASES (continued)

LCOs (continued)

is normally maintained OPERABLE as a backup to the operating RHR loop and satisfies single failure criteria. However, if the standby RHR loop is not OPERABLE, a sufficient alternate method of satisfying single failure criteria is [two] SGs with their water levels ≥ [10]% (wide range indication). Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 in the LCO statement provides for a limited amount of natural circulation heat removal. The period of time allowed is 1 hour because industry operating experience has shown that boric acid stratification is not a problem when natural circulation is limited to 1 hour. Part a of Note 1 prohibits operations that would cause dilution of the RCS boron concentration. Part b of Note 1 requires core outlet temperature to be maintained at least 10°F below saturation and addresses the concern that a bubble might form and restrict flow in the RCS under natural circulation conditions.

Note 2 in the LCO 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 such testing is safe and possible.

Note 3 restricts the start of an RCS pump with an RCS cold leg temperature < [310]°F unless the secondary side water temperature of each SG is ≤ [50]°F above each of the cold leg temperatures . This restriction is to prevent a cold overpressure event due to a thermal transient.

APPLICABILITY

Forced circulation is provided for the reactor coolant to remove decay heat from the core and to provide proper boron homogenization in MODE 5. One loop of RHR provides sufficient circulation for these purposes. The required heat sinks for MODES 1 through 5 are covered by LCO 3.4.4 through LCO 3.4.8.

ACTIONS

A.1 and A.2

If only one RHR loop is OPERABLE, and if < [two] SGs have water level < [10]%, action must immediately be initiated to restore the inoperable RHR loop to OPERABLE status, or action must immediately be initiated to restore the SG levels to satisfy single failure considerations.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If no RHR loop is OPERABLE or in operation, except as provided in Note 1 or Note 2 in the LCO section, all operations involving the reduction of RCS boron concentration must be immediately suspended. Boron dilution requires forced circulation for proper homogenization and the margin to criticality must not be reduced in this type of operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

Verifying that at least [two] SGs are OPERABLE by ensuring their water levels ensures that the single failure criteria is met if the second RHR loop is not OPERABLE. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.7.2

Verifying that at least one RHR pump is in operation ensures that the reactor coolant is in a forced circulation mode and capable of transferring decay heat to the component cooling water via the associated RHR heat exchanger. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.7.3

If all levels in the SGs are ≤ [10]% (wide range indication), then a second RHR loop must be verified OPERABLE. This is because with ≤ [10]% wide range S/G level, the primary tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. Verification that a second RHR loop is OPERABLE ensures that single failure criteria is met. The requirement also ensures that the second RHR loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. The frequency of 7 days ensures that the required flow can be made available and is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

BASES (continued)

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

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 functions of the reactor coolant are: 1) to remove decay heat generated in the fuel, and to transfer this heat to the component cooling water via the Residual Heat Removal (RHR) heat exchangers; and 2) to prevent stratification of the soluble neutron poison, boric acid. The reactor coolant is circulated through [two] RHR loops connected to the reactor vessel, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for both control and protection. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

The RHR System is part of a primary success path and which functions or actuates to 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. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

In MODE 5 with the RCS loops not filled, the LCO requires that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE. One RHR loop with flow through two cold leg injection paths provides sufficient forced circulation to perform the safety functions of the reactor coolant under these unit conditions. The additional RHR loop is required to be OPERABLE to meet single failure considerations.

(continued)

BASES (continued)

LCOs (continued)

Note 1 in the LCO statement allows operation for a limited amount of time without forced circulation. The period of time allowed is 1 hour because operating experience has shown that boric acid stratification is not a problem when forced circulation is secured for limited periods. Part 2 of Note 1 addresses the concern that a bubble might form and restrict flow in the RCS under natural circulation conditions.

Note 2 in the LCO 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.

APPLICABILITY

Forced circulation is provided for the reactor coolant to remove decay heat from the core and to provide proper boron homogenization in MODE 5. One loop of RHR provides sufficient circulation for these purposes. The required heat sinks for MODES 1 through 5 are covered by LCO 3.4.4 through LCO 3.4.8.

ACTIONS

A.1

If only one RHR loop is OPERABLE, action must be initiated immediately to restore the inoperable RHR loop to OPERABLE status to satisfy single failure criteria. The action must continue until RHR loop is restored to OPERABLE status.

B.1 and B.2

If no RHR loop is in operation, except as provided in Note 1 or Note 2 in the LCO section, all operations involving a reduction of RCS boron concentration must be immediately suspended. Boron dilution requires forced circulation for proper homogenization, and the margin to criticality must not be reduced in this type of operation. Also, corrective action must be immediately initiated and continued to restore one RHR loop to an OPERABLE operating status.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

Verifying that at least one RHR pump is in operation ensures that the reactor coolant is in a forced circulation mode and capable of preventing boric acid stratification. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.8.2

Verifying that the second RHR loop is OPERABLE ensures that the single failure criteria is met. The requirement also ensures that the second OPERABLE RHR loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. The frequency of 7 days ensures that the required flow can be made available and is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

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. 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 volume, the heaters, and the heater control and power supplies. 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, respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. The steam bubble limits the volume of non-condensible gases. Relatively small amounts of non-condensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters assures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and assuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The LCO requirement for a steam bubble is reflected implicitly in the safety and accident analyses. 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 non-condensible gases normally present.

The pressurizer is assumed OPERABLE at event initiation, maintaining system pressure and ensuring that a steam bubble exists in the pressurizer. The presence of a steam bubble provides a means of controlling RCS pressure during both steady state operation and normal unit evolutions.

The safety analyses assumes the unit is maintained at nominal operating conditions. Safety and accident analyses do not take credit for pressurizer heater operation. However, an implicit initial condition assumption is that the RCS is operating at conditions of normal pressure, controlled by the pressurizer heaters with no heat loss and resultant pressure decay during the event.

The pressurizer water volume is a process variable that is an initial condition of a Design Basis Accident (DBA) or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 1).

The pressurizer is a component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

The LCO requirement for the pressurizer to be OPERABLE with a water volume $\leq$ [1656] cubic feet, which is equivalent to [92]% of span, ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established 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.

(continued)

BASES (continued)

LCOs (continued)

The requirement for two groups of pressurizer heaters is based in part from the following Reference 2 requirement:

"The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation in MODE 3."

The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability."

LCO 3.3.1, Reactor Trip System Instrumentation, includes a Pressurizer Water Level--High trip, Function 10, which is set at [92]% of span.

APPLICABILITY

The need for pressurizer OPERABILITY is most pertinent in MODES 1, 2, and 3 when the reactor can be made critical and core heat removal must be ensured by subcooled reactor coolant. The steam bubble is required to ensure pressure control during these MODES and for transients that may occur during these MODES. Because the reactor is required to be subcritical in MODES 4, 5 and 6, this LCO is not applicable.

ACTIONS

A.1

Pressurizer level control malfunctions or other unit evolutions may result in a pressurizer level above the nominal upper limit even with the unit at steady state conditions. Nominally the unit will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level--High trip. However, if the unit does not trip, restore pressurizer level below the limit. The Completion Time of 1 hour to perform Required Action A.1 is based on engineering judgment and is reasonable considering the time required to identify the fault and affect repairs or adjust level in the pressurizer.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

The Completion Time of 72 hours to restore 2 groups of pressurizer heaters to OPERABLE status is based on engineering judgment and is reasonable considering the time required to identify the fault and affect repairs.

C.1 and C.2

If Required Action A.1 cannot be completed within the Completion Time, the unit is placed in MODE 3 in 6 hours with reactor trip breakers open and in MODE 4 in the following 6 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train.

D.1 and D.2

The unit must be placed in a MODE in which the LCO does not apply if either one or two groups of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Times of Required Actions B.1. The unit is placed in MODE 3 within 6 hours and in MODE 4 in the following 6 hours. The Completion Time of 6 hours to reach MODE 3 is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This surveillance ensures during steady state operation that pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble by verifying the water volume to be ≤ [1656] cubic feet. The frequency of 12 hours corresponds to verifying the parameter each shift. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through operating experience.

SR 3.4.9.2

The heaters are required to ensure pressure control and the capability of natural circulation in case of loss of offsite power. The [150] kW capacity is based on the heat input to compensate for steady state heat losses from the pressurizer and the heat loss caused by the mini-spray flow used to keep the pressurizer spray nozzle cool. The frequency of 92 days is based on engineering judgment. This frequency has been shown to be acceptable through operating experience.

SR 3.4.9.3

An Emergency Power Supply must be available to the required pressurizer heaters to ensure the OPERABILITY. The frequency of 18 months is consistent with similar verifications of emergency power which is based on Reference 3.

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," Section II.E.3.1, United States Nuclear Regulatory Commission, November 1980.

3. Regulatory Guide 1.3.2, "Criteria for Safety Related Electric Power System for Nuclear Power Plants", United States Nuclear Regulatory Commission, February 1977.

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-activated valves with back pressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system safety limit, [2735] psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self-activating, they are considered independent components. The relief capacity for each valve, [420,000] lbs/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. Adequate protection can be expected from [2] of the [3] installed 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 and/or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5. However for MODE 4, with one or more RCS cold leg temperatures < [310]°F, and MODE 5 overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.16, Cold Overpressure Prevention.

The upper and lower pressure limits are based on the ± 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 that the valves either be set hot or a correlation between hot and cold settings be established. By eliminating the use of safety valves for low temperature overpressure protection, MODE 4 with all RCS cold leg temperatures below [310]°F and MODE 5, there is no potential inconsistency for the lift setting pressure at

(continued)

BASES (continued)

BACKGROUND (continued)

low and high ambient temperature conditions. To permit accurate settings of the safety valves the requirements of LCO 3.0.4 may be suspended in MODE 3 and 4.

The pressurizer safety valves are components which are part of the primary success path and mitigate the effects of postulated accidents. The OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or additional stress analyses required prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) which require safety valve actuation assume operation of [2] pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analyses (Ref. 3) is also based on operation of [2] safety valves. Accidents which 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
d. Loss of Normal Feedwater
e. Loss of All AC Power to Station Auxiliaries

Detailed analyses of the above transients is contained in Reference 2. Safety valve actuation is required in events c, d, and e to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

The pressurizer safety valves are components that are part of the primary success path and which function or actuate to 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. As such, the pressurizer safety valves satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

(continued)

BASES (continued)

LCOs

The pressurizer safety valves are required to open at RCS design pressure and within the ASME specified tolerance to avoid exceeding the design bases, to maintain consistency with accident analyses assumptions, and to comply with ASME Code requirements. Satisfying this LCO helps preserve the RCS pressure boundary, thus avoiding safety consequences and equipment damage.

The notes are to accommodate the testing of safety valves to confirm the lift settings. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires that the valves either be set hot or a correlation between hot and cold settings be established. Provided a cold setting for the safety valves has been performed, the requirements of LCO 3.0.4 may be suspended for up to 18 hours in MODE 3 to permit accurate settings of the safety valves in MODE 3, and in MODE 4 for as long as is required to reach MODE 3 and complete the hot settings of the safety valves.

APPLICABILITY

The LCO is applicable in MODES 1, 2, and 3 because the operability of the safety valves is required to prevent overpressurization of the RCS during potential accidents.

The LCO is applicable in MODE 4 with any RCS cold leg temperature $\geq$ [310]°F to provide RCS overpressure protection until the RCS cold leg temperature reaches the Cold Overpressure Mitigation System (COMS) arming temperature. At temperatures below this point, protection is provided by LCO 3.4.16, Cold Overpressure Mitigation System. OPERABILITY of the safety valves is required to prevent RCS overpressurization during potential accidents.

ACTIONS

A.1

If one safety valve is inoperable, it must be restored to OPERABLE status 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 fission product barrier.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

The unit must be placed in a MODE in which the LCO does not apply if the safety valve cannot be restored to OPERABLE status within the allowed Completion Time. If Required Action A.1 cannot be completed within the Completion Time, the unit is placed in MODE 3 in 6 hours and in MODE 4 with all RCS cold leg temperatures below [310]°F, in the following 6 hours. The Completion Time of 6 hours to reach MODE 3 per Required Action B.1 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems or operators. The Completion Time of 12 hours to reach MODE 4 per Required Action B.2 is sufficient considering conservative cooldown rates. The change to MODE 4, with all RCS cold leg temperatures below [310]°F, is required to reduce potential pressurizer insurges and to accommodate overpressurization without the use of the pressurizer code safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Surveillance Requirements are specified in accordance with the Inservice Inspection and Testing Program. Section XI of the ASME code (Ref. 5) provides the activities and their frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified.

REFERENCES

1. ASME Boiler and Pressure Code Section III, NB-7614.3

2. Watts Bar FSAR, Chapter [15].

3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors", June 1972.

(continued)

BASES (continued)

REFERENCES (continued)

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

5. ASME Boiler and Pressure Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components.

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, ASME code safety valves and PORVs. The PORVs are [air operated] valves which are controlled to open at a specific set pressure when the pressurizer pressure increases and close on decreasing pressurizer pressure. The PORVs may also be manually operated from the control room.

Block valves, 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. As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by the unit 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 performing surveillance on the valves during power operation.

The PORVs, their block valves, and their controls are powered from the vital busses which 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. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The unit has [two] PORVs, each having a relief capacity of [210,000] lbs/hr at [2265] psig. The functional design of the PORVs is based on maintaining pressure to below the Pressurizer Pressure--High reactor trip setpoint following a step reduction of [10]% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer code safety valves and also may be used for cold overpressure prevention. See LCO 3.4.16, Cold Overpressure Mitigation System.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Unit operators employ the PORVs to depressurize the RCS in response to certain unit transients if normal pressurizer spray is not available. For the steam generator tube rupture event, the safety analyses assumes that 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 used in safety analyses for those events which result in increasing RCS pressure for which DNBR criteria is critical. By assuming PORV actuation the primary pressure remains below the high pressurizer pressure trip setpoint, thus the DNBR calculation is more conservative. Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2).

The PORVs are a component that is part of the primary success path and which function to mitigate a design bases accident or transient that assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, the PORVs satisfy Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The LCO ensures that the PORVs and their associated block valves are OPERABLE for manual operation to mitigate the effects associated with a steam generator tube rupture.

By maintaining two PORVs and their associated block valves OPERABLE, single failure criteria is satisfied. The block valves, although not identified as a significant risk by a probabilistic risk assessment, 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.

(continued)

BASES (continued)

APPLICABILITY

The LCO is applicable in MODES 1, 2, and 3. The PORVs will only open automatically when the RCS pressure increases or be required to open manually when it is desired to reduce RCS pressure. The pressure increases only when imbalances exist between the primary heat generated and the heat removal capability of the secondary system. The most rapid pressure increases could only occur during MODES 1 and 2. However because the steam generator is in use in MODE 3, pressure increases challenging the PORVs can also occur. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.16, Cold Overpressure Mitigation System.

The exception to LCO 3.0.4 permits entry into MODES to perform the cycling of the PORVs or block valves to verify their OPERABLE status. This testing is not performed in MODES 4 or 5.

ACTIONS

A.1 and A.2

If the PORV(s) are inoperable due to excessive leakage, presence of flow or temperature downstream of the PORV(s), then it is necessary to either restore the valve(s) within the Completion Time of 1 hour or isolate the flow path by closing the associated block valve(s). The Completion Time of 1 hour is reasonable based on potential challenges to the PORVs and provides the operator adequate time to perform the corrective action. If the Required Actions cannot be met within the Completion Time of 1 hour the unit must be placed in a MODE in which the LCO does not apply as required by Condition D.

B.1, B.2.1, B.2.2, B.2.3, B.2.4, and B.2.5

If one PORV is inoperable, but not due to excessive leakage, it must be either restored or isolated by closing the associated block valve and removing the power to the block valve. The Completion Time of 1 hour is reasonable based on challenges to the PORVs during this time period and provides 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 at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If it cannot be restored within this additional time, the unit must be placed in a MODE in which the LCO does not apply as required by Condition D.

(continued)

BASES (continued)

ACTIONS (continued)

C.1, C.2.1, C.2.2, C.2.3, C.3.1 and C.3.2

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 isolate the flow path by closing the block valve and by removing power to the block valve or by de-activating the associated PORV by removing power to its solenoid valve. The Completion Time of 1 hour is reasonable based on potential challenges to the system during this time period and provides the operator adequate time to correct the situation. Because there is at least one PORV that remains OPERABLE, the operator is permitted an additional Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. If the block valve is restored within the Completion Time of 72 hours the power will be restored and the PORV restored to an OPERABLE status. If it cannot be restored within this additional time, the unit must be placed in a MODE in which the LCO does not apply as required by Condition D.

D.1 and D.2

If the Required Actions of Condition A, B, or C are not met, then the unit must be placed in a MODE in which the LCO does not apply. The unit is placed in at least MODE 3 in 6 hours and in MODE 4 in the next 6 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 6 hours to reach to MODE 4 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.16, Cold Overpressure Prevention.

E.1, E.2.1, E.2.2, E.2.3, and E.2.4

If more than one PORV is inoperable due to causes other than excessive leakage, it is necessary to either restore the valves 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 time is reasonable based on potential challenges to the system during this time and provides the operator adequate time to correct the situation. If the PORV(s) are restored such that only one PORV is inoperable, then the unit will be in Condition B with the time clock started at the original

(continued)

BASES (continued)

ACTIONS (continued)

E.1, E.2.1, E.2.2, E.2.3, and E.2.4 (continued)

declaration of having more than one PORV inoperable. If no PORVs, or an inadequate number of PORVs, are restored within the Completion Time, then the unit must be placed in a MODE in which the LCO does not apply. The unit is placed in at least MODE 3 in 6 hours and in MODE 4 in the next 6 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 6 hours to reach to MODE 4 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.16, Cold Overpressure Prevention.

F.1, F.2.1.1, F.2.1.2, F.2.2, and F.2.3

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour or isolate the flow path by closing and removing power to the block valves or by closing and deactivating the associated PORVs and removing power to their solenoid valves. The Completion Time of 1 hour time is reasonable based on potential challenges to the system during this time and provides the operator adequate time to correct the situation. If the block valves are restored such that only one block valve is inoperable, then the unit will be in Condition C with the time clock started at the original declaration of having more than one block valve inoperable. If no block valves, or an inadequate number of block valves, are restored within the Completion Time, then the unit must be placed in a MODE in which the LCO does not apply. The unit is placed in at least MODE 3 in 6 hours and in MODE 4 in the next 6 hours. The Completion Time of 6 hours is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 6 hours to reach to MODE 4 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.16, Cold Overpressure Prevention.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

The block valve cycling verifies that it can be closed if required during unit operation. The frequency is based on engineering judgment. The frequency of 92 days has been shown to be acceptable based on operating experience. This surveillance is not required if the block valve has been closed with the power removed as part of the Required Actions of this Specification.

SR 3.4.11.2

The CHANNEL CALIBRATION ensures that the PORV setpoint is appropriately established below the Pressurizer Pressure--High reactor trip setpoint to meet functional design requirements . The frequency of 18 months is consistent with the frequencies for similar valves and systems. It is based on performing the calibration during the unit refueling and maintenance outage.

SR 3.4.11.3

Operating the PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of a steam generator tube rupture. The frequency of 18 months is consistent with the frequencies for similar valves and systems. It is based on performing the testing during the unit refueling and maintenance outage

SR 3.4.11.4

The PORVs and block valves must be able to be powered from the emergency power supply and the valves operated through a complete cycle of full travel to ensure their capability to function on a loss of normal power. The 18 month frequency is consistent with similar verifications of emergency power.

(continued)

BASES (continued)

REFERENCES

1. NRC, Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants", February 1977.

2. Watts Bar FSAR, Section [15.2].

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 RCS Operational Leakage

BASES

BACKGROUND

The pressure containing components of the RCS, including the portions of the system out to and including isolation valves, are defined as the Reactor Coolant Pressure Boundary (RCPB). The RCS is comprised of components whose joints are made by welding, bolting, rolling and pressure loading. The RCS is isolated from other systems by valves. Varying amounts of reactor coolant leakage through these valves may occur due to either, normal operational wear or mechanical deterioration. Limits on leakage from the RCPB are required to limit system operation in the presence of excessive leakage. Leakage should be limited to amounts which do not compromise safety. These leakage limits ensure appropriate action can be taken before the integrity of the RCPB is impaired.

This LCO specifies the types and amounts of LEAKAGE which are acceptable during continued plant operation. This LCO is required to protect the RCPB, one of the fission product barriers, against degradation. This ensures that core cooling will be maintained. This LCO is based, in part, on 10 CFR 50 Appendix A, General Design Criteria (GDC) 30 (Quality of Reactor Coolant Pressure Boundary) (Ref. 1). GDC 30 requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods of implementing the requirements of GDC 30 with regard to the selection of leakage detection systems for the RCPB.

The safety significance of leaks from the RCPB can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of reactor coolant LEAKAGE into the containment area is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak occur that is detrimental to the safety of the facility.

(continued)

BASES (continued)

BACKGROUND (continued)

A limited amount of leakage is expected from auxiliary systems within the containment. If leakage occurs from these paths it should be detectable and isolable from the containment atmosphere if possible, so as not to mask any potentially serious RCPB leak.

This LCO protects the RCPB against continuing degradation, thereby protecting the core from inadequate core cooling. It also ensures that the accident analysis radiation release assumptions are not exceeded. The consequences of violating this LCO include the possibility of further degradation of the RCPB which may lead to a Loss of Coolant Accident.

Other related LCOs include 3.4.13, RCS Pressure Isolation Valve Leakage, which specifies leakage limits for certain valves that isolate the high pressure RCS from other low pressure systems. LCO 3.4.14, RCS Leakage Detection Instrumentation, specifies the requirements for the monitoring equipment used to detect leakage into the containment.

APPLICABLE SAFETY ANALYSES

The limits on leakage rates from the RCS are primarily based on the predicted and experimentally observed behavior of pipe cracks. They are also based, to a lesser extent on equipment design leakage and on the sensitivity of the instrumentation for detecting system leakage. The unidentified LEAKAGE limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Based on crack behavior from experimental programs it is estimated that leak rates of hundreds of gpm will precede crack instability. Except for primary-to-secondary leakage, the transient and accident analyses do not address operational leakage. However, operational leakage other than primary-to-secondary is related to the accident analyses for LOCA in that the amount of leakage can affect the probability of such an event. Changes to these leakage limits could affect the margin of safety with respect to a LOCA due to failure of the RCPB.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break accident or, to a lesser extent, other accidents or

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

transients involving secondary steam release to the atmosphere, such as a steam generator tube rupture. Safety analyses for the steam line break accident and the steam generator tube rupture make the assumption of 1 gpm primary-to-secondary leakage as an initial condition of the event. Of the 1 gpm, 0.347 gpm or 500 gallons per day are assumed to be from the defective steam generator. The criteria used to measure the consequences of the steam line break accident are the dose limits of 10 CFR 100 and the Departure From Nucleate Boiling Ratio (DNBR). The steam line break accident and steam generator tube rupture safety analysis show that the limits of this LCO ensure that these criteria are met. The dose consequences resulting from the steam line break accident and the steam generator tube rupture are well within the limits defined in 10 CFR 100 or staff-approved licensing basis (e.g., specified fraction of 10 CFR 100 limits). In addition, the analysis for the steam line break accident shows that no DNB occurs for any of the steam line rupture scenarios.

The primary-to-secondary LEAKAGE limit of this LCO is an initial condition assumed in the accident analyses described above. Changes to these limits could affect the margin of safety associated with these analyses.

Primary-to-secondary leakage is a process variable that is an initial condition of a design basis accident or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 3). In addition, operational leakage is an indication of possible degradation of the RCPB, and therefore also satisfies Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

a. <u>No Pressure Boundary LEAKAGE</u>

No Pressure Boundary LEAKAGE is allowed because it would be 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, possibly leading to gross failure and loss of core cooling.

(continued)

BASES (continued)

LCOs (continued)

b. Unidentified LEAKAGE

One gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring, containment sump level monitoring, can detect within a reasonable time period [(1 hour)]. Violation of this LCO could result in continued degradation of the RCPB, possibly leading to gross failure and loss of core cooling, if the leakage is from the pressure boundary.

c. Identified LEAKAGE

Identified LEAKAGE is defined as leakage into closed systems connected to the RCS that is captured and recovered. Up to 10 gpm of identified LEAKAGE is considered allowable because it provides for leakage from known sources which do not interfere with normal operation and which is well within the capability of the makeup system.

Identified LEAKAGE includes leakage to the containment from sources that are specifically known and located, but does not include pipe or vessel leakage or controlled reactor coolant pump seal leakoff (which is a normal function and is not considered leakage). Violation of this LCO could be indicative of an unexpected amount of leakage from these known sources and therefore indicative of degradation in a component or system. This amount of allowed identified LEAKAGE is also not expected to interfere with the detection of unidentified LEAKAGE.

d. Primary-to-Secondary Leakage

Primary-to-secondary leakage of 1 gpm produces acceptable offsite doses in the steam line break accident analysis. Violation of this LCO could void the offsite dose calculations for this accident. The leakage also shows that a breach of the pressure boundary exists and indicates that further tube deterioration may occur.

(continued)

BASES (continued)

LCOs (continued)

e. Primary-to-Secondary Leakage through One Steam Generator

The 500 gallons per day (gpd) limit on one steam generator is based on the assumption that if a single crack leaked less than this amount, then it would not propagate to a steam generator tube rupture under the stress conditions of a LOCA or a main steam line rupture. If the 500 gpd leaked through many cracks, then they are all very small and the assumptions above are conservative. Violation of this LCO could result in a SGTR event.

APPLICABILITY

The RCS leakage LCOs apply during MODES 1, 2, 3, and 4 because the potential for RCPB leakage is greatest when the RCS is pressurized. Under these conditions, high stresses are applied to the system piping resulting in the potential for crack growth and possible failure of the RCPB. Leakage limits are not provided for MODES 5 and 6 because the reactor coolant pressure is far lower, making leakage less likely and less difficult to control, and because the mechanisms for offsite releases have been reduced or eliminated. Accordingly, the probability and potential consequences of reactor coolant leakage are far lower during MODES 5 and 6.

ACTIONS

A.1 and A.2

If any RCPB leakage is detected, the reactor must be placed in MODE 3 in 6 hours and MODE 5 in the next 30 hours. This action reduces the leakage and also reduces the factors which tend to degrade the pressure boundary. RCPB leakage is non-isolable, indicative of deterioration, and could cause further deterioration resulting in higher leakage, and possibly piping or vessel failure. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cool down in such a time frame on one safety system train. In MODE 5 the pressure stresses acting on the RCPB are much lower and the possibility of further deterioration is not likely.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

With identified LEAKAGE, unidentified LEAKAGE, or primary-to-secondary leakage in excess of the LCO limits, the leakage should be reduced within 4 hours. This Completion Time allows 4 hours to verify leakage rates and either identify unidentified LEAKAGE or reduce whichever type of leakage exists, before the unit must proceed towards shutdown conditions. This action is necessary to prevent further deterioration of the RCPB.

C.1 and C.2

If identified LEAKAGE or unidentified LEAKAGE cannot be reduced to within limits within 4 hours, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The basis for the Completion Times for reaching MODES 3 and 5 are as stated for Required Actions A.1 and A.2 above.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

Verifying that RCS leakage is within the LCO limits assures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by visual inspection. Unidentified LEAKAGE and identified LEAKAGE are demonstrated to be within limits by performance of a 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 feedwater and steam systems. The RCS water inventory balance must be performed with the unit in a reasonably stable state. Therefore, when entering MODES 3 or 4, the requirements of SR 3.0.4 are not applicable for performing an RCS inventory balance.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems which monitor the containment radioactivity. These systems have OPERABILITY requirements established in LCO 3.4.14, RCS Leakage Detection Instrumentation. Also, early warning of identified LEAKAGE through the reactor vessel flange O-rings is provided by the Reactor Head Flange Leakoff

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

System. This system monitors leakage using temperature detectors in the leak-off piping which alarms in the control room. Reactor head flange leakage is directed to the reactor coolant drain tank.

The specified frequency of the RCS water inventory surveillance is based on engineering judgment and operating experience. The frequency permits a reasonable interval for trending of leakage while recognizing the relative importance of early leak detection in the prevention of accidents. The frequency of 72 hours is also justified by the existence of drain tank level alarms along with makeup water operations. Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and the surveillance is not required unless steady state is established. For purposes of leakage determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank level, constant makeup and letdown and reactor coolant pump seal injection and return flows. Pressure boundary LEAKAGE would be detected more quickly by the leakage detection systems referenced in LCO 3.4.14, RCS Leakage Detection Instrumentation.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criteria 30, "Quality Of Reactor Coolant Pressure Boundary.", 1988.

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", United States Nuclear Regulatory Commission, May 1973.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

4. FSAR Chapter 15, "Accident Analysis".

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Pressure Isolation Valve Leakage

BASES

BACKGROUND

The RCS is isolated from other systems by valves. During unit life these interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS Pressure Isolation Valve (PIV) LCO permits system operation in the presence of leakage through these valves in amounts which do not compromise safety. PIV leakage limits apply to leakage rates for individual valves. Leakage from the RCS PIVs is identified LEAKAGE and will be considered as a portion of the total allowed leakage governed by LCO 3.4.12, RCS Operational Leakage.

The basis for this LCO is the 1975 Reactor Safety Study (Ref. 1) which identified potential intersystem Loss Of Coolant Accidents (LOCAs) as a significant contributor to the risk of core melt. A subsequent study (Ref. 2) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leak testing of the PIVs can substantially reduce intersystem LOCA probability.

PIVs are provided to separate the RCS from the following systems :

a. Residual Heat Removal (RHR) System (suction side)
b. Safety Injection (SI) System / accumulators
c. SI System / RHR discharge
d. SI System / SI pump subsystem
e. SI System / charging pump subsystem

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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 1 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the [RHR] System outside of containment. The accident is the result of a postulated failure of the isolation valves in the line which are part of the RCS pressure boundary 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 typically designed for [750] psig, overpressurization failure of the [RHR] low pressure line is postulated. This results in a LOCA outside containment and subsequent risk of core melt.

Reference 2 evaluated various PIV configurations, leak testing of the valves and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leak testing of the PIVs can substantially reduce intersystem LOCA probability.

Leakage from the PIVs is a factor in the dose rates that are used in safety and accident analyses. Therefore the leakage must be maintained within LCO limits to ensure assumptions used in the analyses are valid.

PIV leakage is included as part of identified LEAKAGE covered by LCO 3.4.12, RCS Operational Leakage. RCS operational leakage is a process variable that is an initial condition of a design basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 3).

LCOs

RCS PIV leakage is identified LEAKAGE which is defined as leakage into closed systems connected to the RCS that is captured and recovered. Maximum isolation valve leakage is usually on the order of drops per minute. If leakage increases significantly it suggests that something is operationally wrong and corrective action should be taken as soon as possible. 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.

(continued)

BASES (continued)

LCOs (continued)

In cases where there are two PIVs in series the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leak tested, then one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The LCO leakage limit is based on permitting 0.5 gpm per nominal inch of valve size with a maximum upper limit of 5 gpm (Ref. 4). The previous criterion of 1 gpm for all valve sizes was considered arbitrary and was not an indicator of imminent accelerated deterioration or potential valve failure. A study (Ref. 5) concluded allowable leak rates based on valve size was superior to a single allowable value. The single value imposes an unjustified penalty on the larger valves without providing information on potential valve degradation. In addition, enforcing the single value criteria resulted in higher personnel radiation exposures. The Reference 5 study also concluded that an indexing criterion, similar to Reference 6, be used to account for gross increases in leakages from one test to a later test. This criterion, which has been added to the Surveillance Frequency, provides an indication of valve deterioration.

Leakage tests at lower pressure differentials are permitted by Reference 7. The observed rate will be adjusted to the maximum pressure by assuming leakage is directly proportional to the pressure differential to the one-half power.

APPLICABILITY

The LCO applies during MODES 1, 2, 3 and 4 to ensure that the applicable accident analyses assumptions remain valid and to minimize the probability of intersystem LOCAs. Leakage limits are not provided for MODES 5 and 6 because the RCS pressure has been drastically reduced making leakage less likely and less difficult to control and because the mechanisms for offsite release have been reduced or eliminated. Thus the potential consequences of RCS PIV leakage are far lower during MODES 5 and 6. Leakage from the RCS PIVs is classified as identified LEAKAGE. The leakage must be included as part of the 10 gpm limit specified in LCO 3.3.14, RCS Operational Leakage. Some of the PIVs may also be included as containment isolation valves which are identified in LCO 3.6.6, Containment Isolation Valves.

(continued)

BASES (continued)

ACTIONS

A.1

If RCS PIV leakage in excess of the limits is detected then the leakage must either be reduced or the leakage isolated by closing manual valves or closing and deactivating automatic isolation valves. The Completion Time of 4 hours is based on an NRC evaluation of the probability of a PIV failure coupled with an intersystem LOCA.

B.1 and B.2

If the Required Actions are not completed within the allowed Completion Time the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 6 hours and in MODE 5 within the next 30 hours. This action reduces the leakage and the factors which tend to further degrade the PIVs. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limits and to detect leaking valves as soon as possible. This testing also minimizes the potential for intersystem LOCAs.

The respective specified frequencies are based on the following:

The frequency of 18 months represents the maximum surveillance interval for valves that cannot be exercised during unit operation. This frequency ensures testing if the subsequent requirements are not applicable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.1 (continued)

If the primary system has been depressurized for an extended period and has just been repressurized and the check valves have not been leak tested it is appropriate to confirm that each valve has seated properly before beginning an extended operation at power. The surveillance frequencies of within 72 hours, if testing has not been performed in the last 9 months are based on engineering judgment. The surveillance cannot be performed until RCS pressure is adequate to seat the valves. Therefore, when entering MODES 3 or 4, the requirements of SR 3.0.4 are not applicable for 72 hours until the leak testing of the PIVs can be completed. These frequencies have been shown to be acceptable through industry operating experience.

If the PIV has been actuated or there has been flow through the PIV then the check valve has unseated. It is important to determine how well the valve has reseated prior to beginning operation at power. The surveillance frequency of 24 hours reflects the importance of this requirement and has been shown to be acceptable through operating experience.

SR 3.4.13.2

Verifying that the RHR autoclosure interlock is operable ensures that RCS pressure will not pressurize the RHR system above its design pressure of [750]psig. The interlock setpoint which prevents the valves from being opened is usually set at [380] psig which is the RCS pressure near the top of MODE 4. The frequency of 18 months is based on engineering judgement and the fact that the testing of these interlocks is best performed during a refueling outage. This frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Reactor Safety Study-An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", Appendix V, WASH-1400 (NUREG-75/014), October 1975.

2. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes", NUREG-0677, May 1980.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

4. Safety Evaluation By the Office of Nuclear Reactor Regulation Related to Amendment No. 50 to Facility Operating License No. NPF-2 and Amendment No. 41 to Facility Operating License No. NPF-8, Alabama Power Company, Joseph M. Farley Plant, Unit Nos. 1 and 2, Docket Nos. 50-348 and 50-364, October 15. 1984.

5. EG&G Report, EGG-NTAP-6175, "In Service Leak Testing of Primary Pressure Isolation Valves", R. A. Livingston, February 1983.

6. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants", paragraph IWV-3427(b).

7. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants", paragraph IWV-3423(e).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," (Ref. 1), requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Additional guidance is provided in Regulatory Guide 1.45, (Ref. 2).

A limited amount of leakage is expected from the Reactor Coolant System and from auxiliary systems within the containment. Small amounts of leakage will occur from valve stem packing glands, circulating pump shaft seals, and other equipment that cannot practically be made 100% leaktight. The reactor vessel/closure head seals and safety and relief valves should not leak significantly. However, if leakage occurs, via these paths or via pump and valve seals, it is collected and detected and, to the extent practical, isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leakages are known as identified LEAKAGE and are piped to tanks or sumps so flow rate can be established and monitored during unit operation.

Uncollected leakage to the containment atmosphere from sources such as valve stem packing glands and other sources that are not collected increases the humidity of the containment. The moisture removed from the atmosphere by air coolers together with any associated liquid leakage to the containment is known as unidentified LEAKAGE and is collected in tanks or sumps where the flow rate is established and monitored during unit operation. A small amount of unidentified LEAKAGE may be impractical to eliminate but, it should be reduced to a small flow rate, preferably less than one gpm, to permit the leakage detection systems to positively and rapidly detect a small increase in flow rate. Thus a small unidentified LEAKAGE rate that is of concern will not be masked by a larger acceptable identified LEAKAGE rate.

(continued)

BASES (continued)

BACKGROUND (continued)

It is essential that leakage detection systems have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross pressure boundary failure. Therefore, an early warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow rate changes of from 0.5 to 1.0 gpm can readily be detected in containment sumps by monitoring changes in sump water level, in flow rate or in the operating frequency of pumps. Sumps and tanks used to collect unidentified LEAKAGE and air cooler condensate are instrumented to alarm for increases of from 0.5 to 1.0 gpm in the normal flow rates. This sensitivity provides an acceptable performance for detecting increases in unidentified LEAKAGE.

The reactor coolant contains sources of radioactivity which, 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 become available from the fuel element cladding contamination or defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for air particulate monitoring and of 10^{-6} μCi/cc radioactivity for radiogas monitoring are practical for these leakage detection systems. Radioactivity monitoring systems are included for every unit, both particulate and gaseous activity monitoring, because of their sensitivity and rapid response to leaks from the RCPB.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The safety significance of leaks from the RCPB can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of reactor coolant leakage into the containment area is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to permit corrective action should a leak occur that is detrimental to the safety of the facility. RCS leakage detection instrumentation is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. As such, it satisfies the requirements of Criterion 1 of the NRC Interim Policy Statement (Ref. 3).
LCOs	The only continuous method of protection against RCPB gross failure is the ability of instrumentation to rapidly detect extremely small leaks. This requires that multiple redundant and diverse instrumentation be used to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the unit in a safe condition prior to possible RCPB gross failure.
APPLICABILITY	Leakage Detection Systems are only required to be OPERABLE in MODES 1, 2, 3, and 4 when Reactor Coolant System temperature is greater than 200°F and pressure is greater than atmospheric. These conditions are necessary to have RCPB leakage. With the unit in MODE 5 or 6, the temperature is ≤ 200°F and pressure is maintained low or at atmospheric. Since the design of the RCPB is able to withstand temperatures and pressures far greater than those allowed in MODE 5 or 6, leakage from the RCPB is highly improbable. Therefore, leakage detection in these MODES is not required.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

Should both the containment atmosphere gaseous and particulate radioactivity monitor become inoperable, containment air grab samples may be taken and analyzed for radioactivity every 24 hours to provide periodic information that is adequate to detect RCPB leakage. Provided that at least the containment sump level is OPERABLE and the results of the samples are satisfactory, the unit may continue operation for up to 30 days. The allowance of only 30 days is based on the engineering judgment that extended periods of operation while using alternative monitoring is not prudent since the original design monitoring capability is not being met.

B.1 and B.2.1

If the Containment Floor and Equipment Drain Sump level monitor becomes inoperable, no form of grab sample could provide the equivalent information. However, if continuous monitoring is provided by an OPERABLE containment atmosphere gaseous or particulate monitor and a RCS water inventory balance is satisfactorily performed every 12 hours, then 30 days is allowed to restore the Containment Floor and Equipment Drain Sump level instrumentation.

C.1 and C.2

If Required Actions are not completed within the required Completion Times, the unit must be placed in a MODE where the LCO is not applicable. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Surveillance Requirement 3.4.14.1 is the performance of a CHANNEL CHECK of the containment atmosphere gaseous and particulate radioactivity monitoring instrumentation. A CHANNEL CHECK is a comparison of similar channels, if possible, or the verification by observation that the instrumentation is functioning properly. For channel comparison it is based on the assumption that the two channels of indication should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria it may be an indication that the transmitter or the electronics have drifted outside of their limit. If the channels are within the match criteria it is a reasonable assumption that the channels are within specification with respect to their trip setpoints. The frequency of 12 hours is based on engineering judgment. This frequency has been shown acceptable through industry operating experience.

SR 3.4.14.2

Surveillance Requirement 3.4.14.2 is a performance of a ANALOG CHANNEL OPERATIONAL TEST for the containment atmosphere gaseous or particulate radioactivity monitoring instrumentation. This test is a periodic check of the process control equipment while the unit is at power. When the channel is placed in the test condition the input from the transmitter is removed and the trip output is isolated. This allows a test signal to be introduced into the instrument loop. The input can be measured thus noting the accuracy of the signal conditioning of the process control modules upstream. The trip setpoint can be determined by varying the input and observing the trip status. The frequency of 31 days is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.3

Surveillance Requirement 3.4.14.3 is a performance of a CHANNEL CALIBRATION for all of the leakage detection instrumentation. A CHANNEL CALIBRATION is performed every 18 months, or approximately every refueling. The frequency of 18 months is based on consideration of the magnitude of transmitter drift in the statistical analyses and has been shown to be acceptable through operating experience. This test is a complete check of the process control instrument loop and the transmitter. The transmitter "as found" value is noted and adjustments are made as necessary, with notation of the "as left" value for use in a drift calculation. The transmitter may be calibrated in place by the use of calibrated sources, on a bench using essentially the same type of equipment or be replaced by an equivalent unit calibrated in a laboratory. Completion of this test results in the channel being properly adjusted and expected to remain within the tolerance until the next scheduled surveillance.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, 1988.

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", United States Nuclear Regulatory Commission.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Specific Activity

BASES

BACKGROUND

The purpose of the RCS specific activity LCO is to limit the allowable concentration level of radionuclides in the reactor coolant. The LCO limit is established to minimize the offsite radioactivity dose consequences in the event of a Steam Generator Tube Rupture (SGTR) accident. With a coincident loss of offsite power following a SGTR and reactor trip, the main condenser heat sink is lost through automatic closure of the steam dump valves. Heat removal then occurs by steam release through the main steam safety valves and power-operated relief valves. Opening of these valves establishes a direct path to the environment for release of radioactivity from the steam generator secondary coolant. Under these conditions, with the reactor coolant discharge from the ruptured steam generator tube and an assumed reactor coolant leakage rate of 1 gpm to the steam generator, a much greater fraction of the radioactivity would reach the environment than if the steam were discharged to the condenser.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activities. 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 values. The limiting values for specific activities in the LCO represent standardized limits based upon a parametric evaluation by the NRC of offsite radioactivity dose consequences for typical site locations. These evaluations showed that the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 guideline dose limits, assuming a broad range of site applicable atmospheric dispersion factors in a parametric evaluation. These standard limits on specific activity were also used in establishing standardization in shielding and unit personnel radiation protection practices.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The LCO limitation 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 guidelines (Ref. 1) following a SGTR accident. In the safety analyses (Ref. 2) the specific activity of the reactor coolant is assumed to be at the LCO limits and an existing reactor coolant steam generator tube leakage rate of 1 gpm is assumed. The specific activity of the secondary coolant is assumed to be at its limit of 0.1 μCi/g DOSE EQUIVALENT I-131 in accordance with LCO 3.7.7, Secondary Coolant Specific Activity. The analysis is performed for two cases of reactor coolant specific activity. One case assumes specific activity to be 1.0 μCi/g 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 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity to be 60.0 μCi/g DOSE EQUIVALENT I-131 due to a preaccident iodine spike caused by RCS transients. In both cases the noble gas activity in the reactor coolant is based on a 1% failed fuel assumption. This is closely equivalent to 100/Ē μCi/g of radioactivity. Coincident with the tube rupture a loss of offsite power is also assumed. The reduction in reactor coolant inventory caused by the SGTR initiates a reactor trip due to low pressurizer pressure or overtemperature ΔT trip signals. The assumed coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in steam pressure in the affected steam generator causes steam discharge to the atmosphere through the power-operated relief and main steam safety valves. The unaffected steam generators remove core decay heat by venting steam to the atmosphere until the cooldown is terminated. The safety analysis with the reactor coolant specific activity at the LCO limits and the conservative assumptions applied shows that the radiological consequences of a SGTR accident are within a small fraction of the 10 CFR 100 dose guideline values. Operation with iodine specific activity levels greater than the LCO limit is permissible, provided that the activity levels do not exceed the limits shown in Figure 3.4.15-1 and do not exist for more than 48 hours. The safety analysis, as described above, included both concurrent and preaccident high iodine spiking level analyses which cover up to at least the 60.0 μCi/g DOSE EQUIVALENT I-131 level. The remainder of the above

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

LCO limit permissible iodine levels shown in Figure 3.4.15-1 are acceptable based upon the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of a SGTR accident at these permissible levels could increase the SITE BOUNDARY dose levels, but still be within 10 CFR 100 dose guideline values.

The reactor coolant specific activity is a process variable that is an initial condition of a design basis accident that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The specific activity for iodine is limited to 1.0 μCi/g DOSE EQUIVALENT I-131 and the gross specific activity, excluding iodines, is limited to the number of μCi/g equal to 100 divided by the average beta and gamma disintegration energy, $\bar{E}$. These values represent a reasonable operating capability rather than a specific analytical result for a specified radioactivity dose level at the SITE BOUNDARY. These levels are expected to lead to acceptable 2 hour SITE BOUNDARY doses that are a small fraction of the 10 CFR 100 guideline limit values. The SGTR accident analysis (Ref. 2) 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 a SGTR, lead to SITE BOUNDARY dose limits which exceed the analyzed dose limits established as acceptable in accordance with 10 CFR 100 guideline values.

APPLICABILITY

This LCO is applicable in MODES 1, 2, and 3 with RCS average temperature ≥ 500°F. In MODES 1, 2, and 3, for operations with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and total specific activity are necessary to contain

(continued)

BASES (continued)

APPLICABILITY (continued)

the potential consequences of a SGTR to within the acceptable SITE BOUNDARY dose values. For operation in MODE 3 with RCS average temperature < 500°F and in MODES 4 and 5 the release of radioactivity in the event of a SGTR is prevented since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. In all applicable MODES, with the LCO limits exceeded, an isotopic analysis for iodine concentration is appropriate to monitor the activity level while actions are being taken to reduce the specific activity level.

ACTIONS

A.1

In all LCO applicable MODES, whenever the gross specific activity is > 100/E μCi/g or the DOSE EQUIVALENT I-131 is > 1.0 μCi/g, the Required Action is to sample more frequently and until the activity is restored to within limits. The sample and analysis are to be performed within 4 hours and every 4 hours thereafter until the reactor is no longer in Condition A, i.e. the reactor coolant specific activity returns to within the LCO limits. This action is taken to collect information on the reactor coolant activity to enable appropriate corrective actions and to verify their effect to verify the re-establishment of the specific activity within the LCO limits and to provide the data for NRC reporting requirements (Ref. 4). The Completion Time of 4 hours is reasonable and based upon the typical time necessary to obtain the sample, transport the sample for analysis, and includes an approximate analysis time of 90 minutes.

A.2

When the reactor coolant specific activity is > 100/$\bar{E}$ μCi/g the reactor must be placed in MODE 3 with RCS average temperature < 500°F within 6 hours. The specific activity of 100/E μCi/g is equivalent to about 1% failed fuel, which is an initial condition in the analysis of the postulated

(continued)

BASES

ACTIONS (continued)

A.2 (continued)

SGTR accident. With activity levels of this magnitude present the calculated radiological dose at the SITE BOUNDARY could exceed the required fraction of the 10 CFR 100 guideline values. The change to MODE 3 reactor operation, with RCS average temperature < 500°F, lowers the saturation pressure for the reactor coolant below the set points of the main steam safety valves. This action prevents venting of the steam generator to the environment in the event of a SGTR. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power without challenging safety systems or operators.

B.1

In all LCO applicable MODES, whenever the DOSE EQUIVALENT I-131 is > 1.0 μCi/g, the Required Action is to sample and perform, every 4 hours until the specific activity returns to within limits, a short isotopic analysis of the reactor coolant for iodine, for only I-131, I-133, and I-135. This short analysis can be done more quickly than the full analysis that includes all the radioiodine isotopes, but gives a reasonable estimate of the DOSE EQUIVALENT I-131 concentration. This is true because I-132 and I-134 are very small contributors to the result.

B.2

If the out-of-specification condition was caused by a normal iodine spike, the iodine will peak and decrease to within limits within 48 hours. Therefore, a Completion Time of 48 hours is allowed in Action B.2. If the activity has not decreased an abnormal condition is indicated and the situation reverts to Condition C.

(continued)

BASES (continued)

ACTIONS (continued)

C.1 and C.2

Whenever the reactor coolant specific activity for DOSE EQUIVALENT I-131 exceeds the limits shown on Figure 3.4.15-1 or is > 1.0 μCi/g for a continuous time interval of 48 hours, an abnormal condition is indicated and the reactor must be placed in MODE 3 with RCS average temperature < 500°F within 6 hours. SR 3.4.15.3 is also continued until the specific activity is within limits. The Completion Time of 6 hours is based on engineering judgment and is considered a reasonable time to get to MODE 3 below 500°F without challenging safety systems or the unit operators.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

The surveillance is performed at least once per 72 hours to monitor the gamma isotopic analysis of the reactor coolant. It basically is a quantitative measurement of radionuclides with half lives > 15 minutes, including radioiodines. This measurement considers the sum of the degassed beta-gamma activity and the total of the identified gaseous activities in the sample taken. This surveillance provides an indication of any increase in the specific activity of the reactor coolant. Monitoring of the results of this surveillance allows for proper remedial actions to be taken prior to reaching the LCO limits under normal operating conditions. The surveillance is applicable in MODES 1, 2, and 3 with RCS average temperature ≥ 500°F. The frequency of 72 hours has been shown to be acceptable through operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.2

A radiochemical analysis for $\bar{E}$ determination is required to be performed every 6 months with the unit operating in MODE 1. To assure equilibrium conditions at the time the sample is drawn this surveillance must be performed after a minimum of 2 effective full power days and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for $\geq$ 48 hours. These requirements for $\bar{E}$ determination directly relate to the LCO and are required to verify unit operation within the specified LCO limits. The radiochemical analysis for $\bar{E}$ is a measurement of the average gamma energies per disintegration for isotopes with half lives > 15 minutes, excluding iodines. The frequency of 6 months is based on engineering judgment and the fact that $\bar{E}$ does not change rapidly during operation. This frequency has been shown to be acceptable through operating experience.

SR 3.4.15.3

This surveillance is performed to ensure iodine levels remain within limits following power changes. The frequency, between 2 and 6 hours following a power change of 15% RTP within a 1 hour period, is established as such because iodine spikes occur at this time. Samples at any other time would not provide as accurate results.

REFERENCES

1. Title 10 Code of Federal Regulation, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance", 1973.
2. Watts Bar FSAR, Section [15.6.3]
3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.
4. NRC Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes", September 27, 1985.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Cold Overpressure Mitigation System (COMS)

BASES

BACKGROUND

The purpose of the COMS LCO is to limit reactor coolant pressure at low temperatures to levels which will not compromise reactor pressure boundary integrity by violating 10 CFR Part 50, Appendix G (Ref. 1) Pressure/Temperature (P/T) limits. The reactor vessel is the limiting component for demonstrating that protection is provided. The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. As reactor vessel neutron irradiation accumulates, the vessel material toughness decreases and becomes less resistant to stresses at low temperatures. LCO 3.4.3, RCS Pressure/Temperature Limits, presents requirements for RCS pressure and temperature to be administratively controlled to prevent exceeding the Reference 1 limits.

Regulatory requirements for RCS overpressure prevention are described in NUREG-0800 Revision 2, (Ref. 2), which in turn requires that for low temperature operation of the unit such as startup and shutdown, the overpressure prevention system shall be designed in accordance with the requirements of Branch Technical Position RSB 5-2, (Ref.3).

The OPERABILITY of two Power Operated Relief Valves (PORVs), or a RCS vent, all capable of relieving RCS pressure, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 during low temperature RCS operation. Either PORV valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle Reactor Coolant Pump (RCP) with the secondary water temperature of the steam generator $\leq$ 50°F above the RCS cold leg temperature, or (2) the start of [1] charging pump[s] and subsequent injection into a water-solid RCS.

(continued)

BASES (continued)

BACKGROUND (continued)

PORV Requirements

As designed for the COMS, each PORV is signaled to open should the RCS approach a pressure limit as determined by the COMS actuation logic. The function of the cold-overpressure actuation logic is to monitor both the RCS temperature and pressure to ascertain when an unacceptable condition regarding the Ref. 1 limits is being approached. The wide-range RCS temperature indications are auctioneered to select the lowest temperature signal. The low signal is processed through a function generator which calculates a pressure limit for the prevailing temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide-range pressure channel and if the indicated pressure meets or exceeds the calculated value, a PORV will be signaled to open.

The maximum allowable P/T actuation logic setpoints are provided in Figure 3.4.16-1. The open setpoints of the PORVs are normally staggered so that only one valve would open during any cold overpressure transient. Having the setpoints of both valves within the limits of Figure 3.4.16-1 ensures that the Ref. 1 limits will not be exceeded for the events analyzed.

When a PORV is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed. The system pressure then decreases, as the relief valve releases coolant, 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 (continued)

BACKGROUND (continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent of sufficient size exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in the event of an RCS overpressure transient. The only possibility of an overpressure event occurring would be if the relieving requirements of a transient exceeded the capabilities of the vent. Thus it must be verified that the vent path is capable of relieving the flow resulting from the limiting mass or heat input transient while maintaining pressure below the Ref. 1 P/T limits. The required vent capacity may be provided by one or more vent paths.

In addition to opening an RCS vent to meet the flow capacity requirement, it is acceptable to remove a pressurizer code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS accordingly. Also, the only vent paths that can be utilized for venting the RCS are those which are above the level of reactor coolant which would not result in continuous draining of the RCS when opened.

The consequence of not having an OPERABLE COMS is that the Ref. 1 P/T limits could be exceeded if the RCS were to be subjected to a cold overpressure transient. The potential for overpressurization is most acute when the RCS is water solid and pressure fluctuations could occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could lead to brittle cracking of the reactor vessel.

APPLICABLE SAFETY ANALYSES

Analyses have been performed to show that the reactor vessel is adequately protected against exceeding the P/T limits dictated by Ref. 1. In MODES 1, 2 and 3 with RCS temperature $\geq$ [310]°F), the pressurizer safety valves will prevent RCS pressure from exceeding the Ref. 1 limits. However, at a nominal temperature of [310]°F and below, the safety valves no longer provide protection from exceeding the Ref. 1 limits. Below this temperature, alternate means

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

of overpressure prevention must be made available. The actual temperature at which the pressure in the Ref. 1 P/T limit curve falls below the pressurizer safety valve setpoint of [2485] psig varies with time as reactor vessel material toughness decreases due to neutron embrittlement. Thus, each time the Ref. 1 P/T Limit Curves are revised, the COMS must be reevaluated to ensure that their functional requirements can be satisfied using the PORVs.

Transients which are capable of over-pressurizing the RCS are categorized as either mass input or heat input transients. Some examples are as follows:

<u>Mass input type transients</u>

1. Inadvertent safety injection
2. Charging/letdown flow mismatch

<u>Heat input type transients</u>

1. Inadvertent actuation of pressurizer heaters
2. Loss of residual heat removal cooling
3. RCP startup with temperature asymmetry within the RCS or between the RCS and steam generators

To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require for MODES 4, 5, and 6: lockout of all but [1] charging pump and all safety injection pumps, blocking safety injection actuation circuits, immobilizing accumulator discharge isolation valves, and disallowing start of an RCP if secondary temperature is more than [50]°F above primary temperature in any one loop.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

Maximum allowed PORV P/T setpoints (shown in Figure 3.4.16-1) are derived by analyses which model the performance of the system assuming the design basis mass input and heat input transients. These analyses take into consideration pressure overshoot and undershoot beyond the PORV open and close setpoints, which can occur as a result of time delays in signal processing and valve stroke times. Operation with setpoints less than or equal to the derived setpoints ensures that the Ref. 1 P/T limits will not be violated.

The Ref. 1 P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement. Revised limits are determined using both neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. These examinations are performed as required by Ref. 4 and as discussed in Bases 3.4.3, RCS Pressure/Temperature Limits. The maximum allowed PORV setpoints from Figure 3.4.16-1 will be updated when the revised P/T limits conflict with the COMS analysis limits.

The PORVs are considered active components. Thus the failure of one OPERABLE PORV must be assumed if it represents the worst case single active failure.

RCS Vent Performance

With the RCS depressurized, an open vent of [3] square inches, is proven by analysis to be capable of mitigating a cold overpressure transient. This is accomplished by proving that the vent capacity is greater than the flow resulting from the limiting design basis transient at pressures less than the minimum RCS pressure on the P/T Limit Curve.

The RCS vent is a passive component, and therefore is not subject to active failure.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Selection Criteria

The COMS controls pressure versus temperature and, therefore, controls a process variable that is an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 6).

LCOs

This LCO is required to limit RCS pressure at low temperatures to within the Ref. 1 P/T limits. Violation of this LCO could lead to the loss of cold overpressure mitigation and violation of the Ref. 1 limits as a result of an operational transient.

There are 2 elements of the LCO which allow for cold overpressure mitigation.

1. Two PORVs with appropriate lift settings as provided in Figure 3.4.16-1,

2. The RCS depressurized with an open vent of at least [3] in^2.

Each of these four methods of cold overpressure prevention meet the criteria of Reference 3 and each is capable of mitigating RCS cold overpressure transients.

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is less than or equal to [310]°F, MODE 5, and MODE 6 with the reactor vessel head on. Above [310]°F, overpressure protection from the pressurizer safety valves is adequate for meeting the Ref. 1 P/T limits. With the reactor vessel head off, there is no need for overpressure prevention.

LCO 3.4.3, RCS Pressure/Temperature Limits, provides the operational limits for pressure and temperature for all MODES. LCO 3.4.10, Pressurizer Safety Valves, requires the OPERABILITY of the pressurizer safety valves which provide overpressure protection during MODES 1, 2, and 3. This LCO requires the OPERABILITY of a COMS in MODES 4, 5, and 6.

Low temperature overpressure prevention is most critical when the RCS is water solid, and any mass input or heat input transient may cause a very rapid increase in RCS pressure, such that little or no time is available for operator action to mitigate the event.

ACTIONS

A.1

With one PORV inoperable, it must be returned to OPERABLE within 7 days. The Completion Time of 7 days is based on engineering judgment considering the fact that only one of the subject relief valves is required to mitigate an overpressure transient. Also, the likelihood of an active failure during this time period is very low.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

When all PORVs are inoperable, the RCS should be depressurized per Required Action B.1, and a vent established within a Completion Time of 8 hours. The vent should have size of ≥ [3] square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. Required Action B.1 is appropriate because Ref. 3 requires that the COMS should be able to perform its function assuming any single active component failure. Thus, Required Action C.1 will protect the Reactor Coolant Pressure Boundary (RCPB) from transients which could lead to a cold overpressure event and possibly brittle failure of the reactor vessel. The Completion Time of 8 hours to depressurize and vent the RCS is based on engineering judgment. During this time period the probability of an overpressure event will be relatively low due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

All but [1] centrifugal charging pump are verified locked out (power removed) to minimize the potential for a Cold Overpressure event. The Frequency of once per 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.16.2

The RHR suction relief valve shall be demonstrated to be OPERABLE by verifying that their respective RHR suction isolation valve is open. The isolation valves are verified to be open with the relative high frequency of every 12 hours in order to minimize the likelihood of accidental closure. The Frequency of once per 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

The RCS vent is proven to be OPERABLE by verifying its open condition either:

a. once per 12 hours for open manual valves, or

b. once every 31 days when the vent pathway is provided with a valve which is locked, sealed, secured in the open position or removed altogether.

The passive vent arrangement needs only to be open to be OPERABLE. For valves which can be closed, their position must be verified with a higher frequency than those secured in the open position. Thus the position of the unlocked valves must be verified every 12 hours to assure of no accidental closure. The surveillance interval of 31 days for verifying the secured vent path open is consistent with other surveillances which verify condition the of secured components. This surveillance need only be performed if the vent is being used to satisfy the requirements of this LCO. The frequencies stated above have been shown to be acceptable through industry operating experience.

SR 3.4.16.4

The PORV block valves must be open for the corresponding PORVs to be OPERABLE. Because these motor operated block valves are controlled remotely, they must be verified to be open every 72 hours. This surveillance need only be performed if the PORV is being used to satisfy the requirements of this LCO. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.16.5

The RHR suction isolation valves must be verified open with power to the valve operator removed to ensure no inadvertant action causes the valve to go closed which would cause the RCS to be isolated from the RHR.

(continued)

BASES (continued)

SURVEILLANCE REQUIRMENTS (continued)

SR 3.4.16.6

Performance of an ANALOG CHANNEL OPERATIONAL TEST is required every 31 days to verify and adjust, as necessary, the PORV open setpoints. This test will verify on a monthly basis that the PORV lift setpoints are within the maximum allowed per Figure 3.4.16-1. This frequency has been shown to be acceptable through industry operating experience.

SR 3.4.16.7

Performance of a CHANNEL CALIBRATION on the PORV actuation channel is required every 18 months to adjust the whole channel such that it responds within the required range and accuracy to known input. This frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."
2. NUREG-0800, "USNRC Standard Review Plan," Rev. 2, July 1981
3. Branch Technical Position RSB-5.2, "Overpressurization Protection Of Pressurized Water Reactors While Operating At Low Temperatures".
4. Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements".
5. ANSI/ANS-58.9-1981, "Single Failure Criteria For Light Water Reactor Safety-Related Fluid Systems".
6. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Loops - Test Exceptions

BASES

BACKGROUND

The primary purpose of this Test Exception is to provide an exemption to LCO 3.4.4, RCS Loops - Modes 1 and 2, to allow certain PHYSICS TESTS (Natural Circulation Cooldown and Boron Mixing) to be performed. Section XI of 10 CFR Part 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 occurences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in General Design Criterion 1, Quality Standards and Records, (Ref. 2)

The key objectives of a test program are: to provide assurance that the facility has been adequately designed; to validate the analytical models used in the design and analysis; to verify the assumptions used to predict unit response; to provide assurance that installation of equipment at the facility has been accomplished in accordance with the design; and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

The tests include verifying the ability to establish and maintain natural circulation following a unit trip from 10% to 20% of RATED THERMAL POWER, performing natural circulation cooldown on emergency power, and during the cooldown showing that adequate boron mixing occurs and that pressure can be controlled using auxiliary spray and pressurizer heaters powered from the emergency power sources.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The tests described above require operating the unit without forced convection flow and as such are not bound by any safety analyses. This test exception was not evaluated against the Selection Criteria of the NRC Interim Policy Statement (Ref. 3). Reference 3 allows the test exceptions to be included as part of the LCOs which they affect. It was retained as a separate LCO for clarity and is appropriately cross referenced.
LCOs	This LCO provides an exemption to the requirements of LCO 3.4.4, RCS Loops - Modes 1 and 2. The exemption is allowed even though there are no bounding safety analyses. However, these tests are allowed, since they are performed under close supervision during the test program and provide valuable information on the units capability to cooldown without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This is established to provide a heat input from nuclear heat without exceeding the natural circulation heat removal capabilities. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, MODE 1 Physics Tests Exceptions.
ACTIONS	<u>A.1</u> When THERMAL POWER is > the P-7 interlock setpoint [10 %], the only acceptable action is to ensure the reactor trip breakers are open immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the reactor trip breakers will shutdown the reactor and prevent operation of the fuel outside of its design limits.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

Verification that the power level is < the P-7 interlock setpoint [10%] will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Unit operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.17.2

The power range and intermediate range neutron detectors and the P-7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. An ANALOG CHANNEL OPERATIONAL TEST is performed within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the Reactor Trip System is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The time limit of 12 hours is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants", 1988.

3. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the Emergency Core Cooling System (ECCS) accumulators are to supply water to the reactor vessel during the blowdown phase of a 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 continue to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

The refill phase of a LOCA follows immediately wherein reactor coolant inventory has vacated the core through steam flashing and ejection out the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with addition of safety injection water.

The accumulators are passive components since no operator or control actions are required 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.

Each accumulator is piped into a 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 are interlocked by P-11, with the pressurizer pressure measurement channels to ensure the valves will automatically open as RCS pressure increases to above the P-11 setpoint. The interlock also prevents inadvertent closure of the valve during normal operation prior to an accident. The valves will, however, automatically open as a result of a Safety Injection (SI) signal. These features

(continued)

BASES (continued)

BACKGROUND (continued)

ensure the valves meet the requirements of IEEE Standard 279-1971, (Ref. 1), for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume and nitrogen gas pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three accumulators are 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.

This LCO helps to ensure that the following acceptance criteria established by 10 CFR 50.46 (Ref. 2), for the ECCS will be met following a LOCA:

a. Maximum fuel element cladding temperature of less than 2200°F.

b. Maximum cladding oxidation of less than 0.17 times the total cladding thickness before oxidation.

c. Maximum hydrogen generation from a zirconium-water reaction of less than 0.01 times the hypothetical amount of that which 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.

d. The core is maintained in a coolable geometry.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 3) and in the large Steam Line Break (SLB) analysis.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The loss of offsite power assumption is required by regulation (Ref. 4) and conservatively imposes a delay wherein the ECCS pumps

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

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 is assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump. During this event the accumulators discharge to the RCS as soon as RCS pressure decreases 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 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. The delay time is conservatively set with an additional [2.86] seconds to account for SI signal generation.

The worst case small break LOCA analysis also assumes a time delay before pumped flow reaches the core. For the larger range of small breaks, between [6] inch and [4] inch equivalent diameter, the rate of blowdown is such that the increase in fuel clad temperature is terminated primarily by the accumulators with pumped flow then providing continued cooling. As break size decreases,to [3] inches eqivalent diameter, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, <[2] inch equivalent diameter, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

For both the large and small break LOCA analysis, 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. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling during the core reflooding portion of the transient. The analysis takes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

accumulator water volume setpoints are determined based on an allowable variation about the analysis value.

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 RCS concentration for post-LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg - hot leg recirculation switchover time and minimum sump pH.

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 effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Ref. 3).

The accumulator pressure, boron concentration and volume are process variables that are initial conditions of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such they satisfy the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 5).

In addition, the accumulators are components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such they satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5).

(continued)

BASES (continued)

LCOs

The LCO ensures the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure 100% of the contents of three of the accumulators will reach the core during the event. This is consistent with the analysis assumption that the contents of one accumulator spills through the break. If less than three accumulators are injected during the blowdown phase of a LOCA the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. The limits on accumulator volume, boron concentration, nitrogen cover-pressure, and isolation valve fully open ensures that the accumulators will be OPERABLE and adequately perform their function.

A required boron concentration consistent with the refueling water storage tank requirements ensures no loss of SHUTDOWN MARGIN would occur if an accumulator was to inject while shutdown. In the LOCA analysis, the accumulator's function is to provide RCS makeup and core cooling until the ECCS pumps can start and deliver flow to the RCS.

The accumulator isolation valves are not single failure proof. Therefore, in order to ensure accumulator OPERABILITY, whenever the valves are open, power shall be removed from these valves.

APPLICABILITY

MODES 1, 2, and MODE 3 with RCS pressure $\geq$ [1000] psig: The accumulator OPERABILITY requirements are based on full power operation. The contents provide cooling water to the reactor vessel in the event of a LOCA during the time period of emergency diesel startup, loading and the associated delay in obtaining ECCS pump flow.

This LCO is only applicable at pressures $\geq$ [1000] psig. Below [1000] psig, the rate of RCS blowdown is such that the safety injection pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2), limit of 2200°F for LOCA breaks less than [6] inches in equivalent diameter.

MODE 3 < [1000] psig and MODES 4 to 6: Under these conditions 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.

(continued)

BASES (continued)

ACTIONS

A.1

The ability of the accumulators to perform their core cooling functions during a LOCA is not dependent on the boron concentration of the injected water. Therefore, a 72 hour Completion Time is acceptable to restore the boron concentration of an accumulator.

B.1

With an accumulator inoperable for reasons other than a closed isolation valve or boron concentration, the required contents of the inoperable accumulator cannot be assumed to reach the core during a LOCA. Due to the severtity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open and remove power to the valve, restore the proper water volume or nitrogen pressure, ensures prompt action is taken to return the inoperable accumulator to OPERABLE status.

C.1 and C.2

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. Six hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power conditions without challenging unit systems or operators. Any further reduction to MODE 4 or lower is not required by this LCO as long as pressurizer pressure is maintained lower than the LCO limit.

D.1

With more than one accumulator inoperable, sufficient contents cannot be assured to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur under these conditions, LCO 3.0.3 must be entered immediately to place the unit in a mode where the LCO is not applicable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification of valve position ensures 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 can be reduced. A frequency of once per shift, 12 hours, has been established, and has been shown to be acceptable through industry operating experience.

SR 3.5.1.2 and 3.5.1.3

These SRs ensure that the nitrogen cover-pressure and the borated water volume contained in the accumulators are sufficient to ensure adequate injection during a LOCA to prevent exceeding the 10 CFR 50.46 limit of 2200°F clad temperature. A frequency of once per shift, 12 hours, has been established, based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.5.1.4

This SR ensures accumulator boron concentration is within the required limits. A 31 day frequency is adequate to identify changes which could occur from mechanisms such as in-leakage. Sampling once within 6 hours following accumulator volume changes assures that leakage from the RCS will not dilute the accumulator boron concentration below the allowable limit. This surveillance is not required within 6 hours after volume changes on the affected accumulator if the makeup is from the RWST since it has a more restrictive boron concentration band specified in LCO 3.5.4.

SR 3.5.1.5

This SR ensures that an active failure could not result in the closure of an accumulator motor-operated isolation valve coincident with a LOCA. The 31 day frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. IEEE Standard 279, Revision 1971, "Criteria For Protection Systems for Nuclear Power Generating Stations", April 5, 1972.

2. Title 10 Code of Federal Regulations, Part 50.46, "Acceptance criteria for Emergency Core Cooling Systems for light water nuclear power plants", 1974.

3. WATTS BAR FSAR Sections 6 and 15.

4. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants, No. 35 Emergency Core Cooling, No. 36 Inspection of Emergency Core Cooling System, No. 37 Testing of Emergency Core Cooling System", 1988.

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.2 ECCS Trains - Operating ($T_{avg} \geq 350°F$)

BASES

BACKGROUND

The function of the Emergency Core Cooling System (ECCS) is to provide core cooling and negative reactivity to ensure the reactor core is protected after any of the following accidents:

a. Loss Of Coolant Accident (LOCA), 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.

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 three phases of ECCS operation; injection, cold leg recirculation and hot leg 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. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required NPSH to the ECCS pumps, suction is switched to the containment sumps. After approximately [15] hours the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and the resulting boron precipitation.

The ECCS consists of two redundant 100% capacity trains consisting of three separate subsystems: i.e. centrifugal charging, high head; safety injection, intermediate head; and residual heat removal, low head. The ECCS accumulators and the RWST are also part of the ECCS.

(continued)

BASES (continued)

BACKGROUND (continued)

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem from a common RWST supply header. The discharge from the centrifugal charging pumps combine and then divide again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the Safety Injection (SI) and Residual Heat Removal (RHR) pumps divide and feed an injection line to each of the RCS cold legs. Throttle valves are set to balance the flow to the RCS. This flow balance assures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period the Steam Generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump and the RHR heat exchangers are placed in service. The RHR pumps then supply the other ECCS pumps. Initially recirculation is through the same paths as the injection phase except that the RHR cross-tie valves must be closed to ensure train separation. Subsequently, recirculation alternates injection between the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as Steam Line Breaks (SLBs). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at end of each cycle.

This LCO ensures that peak clad temperature remains below the 10 CFR 50 Appendix K limit of 2200°F thereby precluding fuel melting, maintaining a coolable core geometry and limiting the cladding metal-water reaction. It also ensures that long term cooling is available following a LOCA, limits the magnitude for a post-trip return to power following a SLB to acceptable levels, and ensures that containment temperature limits are met. The active ECCS trains, along with the passive accumulators and the RWST covered in LCO 3.5.1 and LCO 3.5.4 respectively, provide the cooling water necessary to meet GDC 35 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break LOCA event at full power (Ref. 2). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps are credited in the small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and SLB events also credit the centrifugal charging pumps.

The effects on containment mass and energy releases are accounted for in the appropriate analyses (Ref. 2).

A large break LOCA event, with a loss of offsite power and a single failure disabling one train of ECCS, establishes the OPERABILITY requirements for the ECCS. As power decreases, and in lower MODES, the consequences of a large break LOCA are less severe and other events may control OPERABILITY requirements as described in the Applicability section.

This LCO ensures ECCS flow will be available to deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. During this period the steam generators continue to serve as the heat sink, providing part of the required core cooling.

For a loss of offsite power, the ECCS pumps are assumed to be loaded on the emergency diesel generator shortly after the diesel generator's connection to the safeguards bus, which occurs within [12.86] seconds, [2.86] seconds for signal generation and [10] seconds for the diesel to reach full speed, following SI signal generation. After loading, the centrifugal charging pump starts and reaches its rated speed in [7.5] seconds. The SI pump starts and reaches its rated speed in [11]seconds. The RHR pump starts and reaches its rated speed in [16] seconds. However, the ECCS pumps are not credited at this time, for any event. For a large LOCA that is conincident with a loss of offsite power, ECCS pump flow is not credited until completion of the loading sequence, [32.86] seconds after reaching an actuation setpoint in the analysis. For smaller break LOCAs and other events, SI pump flow is not credited until RCS pressure drops below the pump's shutoff head.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The ECCS trains are systems and components that are part of the primary success path and which function or actuate to 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. As such they satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The LCO establishes the minimum equipment required to be available to accomplish the core cooling safety function following accidents which render the steam generators effectively unavailable, such as a large LOCA. Two independent and redundant ECCS trains are required to ensure sufficient ECCS flow is available assuming a single failure somewhere in either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump, and to supply its flow to the RCS hot and cold legs.

The requirements of this LCO are derived principally from events involving a LOCA, and particularly the 10 CFR 50 Appendix K (Ref. 4) evaluation of the maximum hypothetical accident. Failure to meet these requirements could result in the inability to match core heat generation, leading to fuel melting, increased clad metal-water reaction, and potential for alteration of core geometry for these low probability events (Ref. 5).

(continued)

BASES (continued)

APPLICABILITY

Although the ECCS train OPERABILITY requirements are based on full power operation, the ECCS OPERABILITY requirements are applicable in MODES 1, 2 and 3. This is because the safety analysis assumes the respective accidents credible in these MODES and does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pumps performance is also based on the small break LOCA and the SLB which establishes the pump's performance curve and has less dependence on power.

This LCO is only applicable at MODE 3 and above when the highest T_{avg} channel is $\geq$ 350°F.

As indicated in Note 1, the flow path may be isolated for a short period, under controlled conditions, to perform tests ensuring the continued availability of the flow path. The flow path is readily restorable from the control room, and the benefits of ensuring the OPERABILITY of the flow path outweigh the decreased availability during this period.

Note 2 provides a means to determine that ECCS pumps are OPERABLE if they have been made inoperable in the lower Modes in accordance with LCO 3.4.16, Cold Overpressure Prevention.

ACTIONS

A.1

This 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 component in a train does not render the ECCS incapable of performing its function, provided 100% of the safety injection flow equivalent to a single train is available. Neither does the inoperability of two different components, each in a different train, necessarily make both ECCS trains inoperable. This condition allows some increased flexibility in unit operations under circumstances that do not render both ECCS trains incapable of performing their design function.

(continued)

BASES (continued)

ACTIONS (continued)

A.1 (continued)

An event accompanied by a loss of offsite power and the failure of an emergency diesel generator can disable one ECCS train until power is restored. Reliability analysis (Ref. 6) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

Examples of acceptable combinations of out of service components include, but are not limited to:

a. Centrifugal charging pump in train A and SI pump in train B. Components serve different functions.

b. Centrifugal charging pump in train A and containment sump isolation valve in train B.

Examples of unacceptable combinations of inoperable components include, but are not limited to:

a. Centrifugal charging pump in train A and centrifugal charging control valve in train B failed closed. Prevents charging flow to one cold leg.

Reference 7 describes situations in which one component, such as a RHR crossover valve, can disable both ECCS trains causing a loss of function and requiring entrance into LCO 3.0.3.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. Six hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power conditions without challenging unit systems or operators.

B.2

Continuing the unit shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience, to reach conditions where this LCO is no longer applicable without challenging unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

This surveillance ensures that the flow path from the ECCS pumps to the RCS is properly aligned. These valves can disable the function of both ECCS trains, invalidating the accident analyses. As misalignment of these valves could render both ECCS trains inoperable, a frequency of 12 hours is appropriate.

SR 3.5.2.2

This surveillance ensures that the flow path from the RWST to the RCS is properly aligned by requiring a verification of the lineup of those valves which could be inadvertently repositioned. Failure of one of these valves will affect only one ECCS train. A monthly frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode when the unit is operating in MODES 1, 2 or 3. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This also will prevent water hammer, pump cavitation and compressible gas; e.g., air, nitrogen, or hydrogen, from being pumped into the reactor vessel following an SI signal or during shutdown cooling. A monthly frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

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 only one point of the pump characteristic curve. This verifies that both 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. Surveillance Requirements are specified in the Inservice Inspection and Testing Program and provides the activities and frequencies necessary to satisfy the requirements, but allowable pump degradation is based on FSAR accident analysis assumptions, not ASME Section XI. No additional requirements are specified.

SR 3.5.2.5

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. The high pressure injection valves have mechanical stops or inline flow restricting orifices to position them properly to restrict flow to a ruptured cold leg. These actions ensure that the other cold legs receive at least the required minimum flow following a LOCA. The proper valve alignment for the recirculation phase is also verified by this SR. An 18 month frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.6

The ECCS throttle valves are designed to ensure that the required flow reaches each of the RCS cold legs. This SR ensures that their stops are positioned correctly on a regular periodic basis. An 18 month frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

SR 3.5.2.7

Each of the ECCS pumps in this SR are required for proper ECCS operation. This SR demonstrates that they will start on an SI signal when required. A frequency of 18 months has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

SR 3.5.2.8

After a DBA that empties the RWST, the containment sump is relied upon to collect water and be recirculated back through the ECCS pumps to the RCS. To ensure that the pumps are not damaged by debris from containment, screens are provided to filter trash from getting in the racks. This SR verifies that no debris has collected in the sump or on the screen. A frequency of 18 months has been established to conincide with a unit returning to service from outage where debris may have accumulated due to activities in containment.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants, No. 35 Emergency Core Cooling, No. 36 Inspection of Emergency Core Cooling System, No. 37 Testing of Emergency Core Cooling System", 1988.

2. WATTS BAR FSAR Chapters 6 and 15.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

4. Title 10 Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models", 1987.

5. Title 10 Code of Federal Regulations, Part 50.46, "Acceptance criteria for Emergency Core Cooling Systems for light water nuclear power plants", 1974.

6. NRC memorandum R. L. Bayer to V. Stello, Jr., Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

7. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.3 ECCS Trains - Shutdown (T_{avg} < 350°F)

BASES

BACKGROUND	The Background section for Bases 3.5.2 is applicable to this Bases.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases 3.5.2 is applicable to this Bases.
LCOs	The LCO section of Bases 3.5.2 is applicable to this Bases.
APPLICABILITY	MODE 4: The ECCS train OPERABILITY requirements are based upon providing adequate core cooling in the event of a LOCA. With the 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. Increased RCS pressure or power levels would require increased cooling capacity and single failure consideration. MODES 1, 2 and 3: These conditions are covered by LCO 3.5.2.
ACTIONS	<u>A.1</u> With no ECCS train OPERABLE, due to the inoperability of the RHR pumps, heat exchanger, or flow path, the unit is not prepared for the low pressure, high volume response to design basis events requiring Safety Injection (SI). Immediate actions which would restore at least one ECCS train to OPERABLE status, ensures prompt action is taken to restore the required cooling capacity. With one RHR pump and heat exchanger inoperable, it would be unwise to require the unit to go to MODE 5, where the only available heat

(continued)

BASES (continued)

ACTIONS (continued)

A.1 (continued)

removal system is the RHR, and a single failure might disable this path. Therefore, the appropriate action is to initiate actions to restore one ECCS train and to continue the actions until the train is restored to OPERABLE status.

B.1

With the required centrifugal charging pump or flow path inoperable, the unit is not prepared to provide high pressure response to design basis events requiring Safety Injection (SI). The 1 hour Completion Time to restore at least one ECCS train to OPERABLE status ensures prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit in MODE 5 where an ECCS train is not required.

C.1

When the Required Action and associated Completion Time of B.1 cannot be met, a controlled cooldown should be commenced. This action is only required if at least one train of RHR is OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The surveillances of Bases 3.5.2 are applicable to this SR.

REFERENCES

The references of Bases 3.5.2 are applicable to this Bases.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The Refueling Water Storage Tank (RWST) supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refuelings and to the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) during accident conditions.

The RWST supplies both trains of the ECCS and the CSS during the injection phase of LOCA recovery. A motor-operated isolation valve is provided to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST Low-Low-level signal. Use of a single RWST to supply both trains of ECCS and CSS is acceptable since the RWST is a passive component and passive failures are not a required assumption in the analysis of design basis events.

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS Volume Control Tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves are interlocked such that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The affects of this delay are discussed in the Applicable Safety Analyses section of this Bases.

During normal operation in MODES 1, 2 and 3, the Safety Injection (SI) and Residual Heat Removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and CSS pumps are provided with recirculation lines which ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

(continued)

BASES (continued)

BACKGROUND (continued)

When the suction for the ECCS and CSS pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST. If not isolated, this could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that the RWST contains sufficient borated water to support the ECCS during the injection phase, ensures a sufficient water volume exists in the containment sump to support continued operation of the ECCS and CSS pumps at the time of transfer to the recirculation mode of cooling, and ensures the reactor remains subcritical following a LOCA. Insufficient water inventory in the RWST could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in loss of shutdown margin or excessive boric acid precipitation in the core following a LOCA as well as excessive caustic stress corrosion of mechanical components and systems inside containment.

The RWST also has requirements placed upon the volume, boron concentration and temperature for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different than the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the Steam Line Break (SLB) analysis for ensuring the required shutdown capability and in the inadvertent ECCS analysis although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The maximum temperature is a conservative assumption which minimizes the additional cooling in the feedline break event analysis and the minimum temperature is an assumption in both the SLB and inadvertent ECCS analyses although the inadvertent ECCS event is typically non-limiting.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

During accident conditions the RWST provides a source of borated water to the ECCS and CSS pumps. As such it provides containment cooling and depressurization, core cooling, replacement inventory, and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analysis concerning each of these systems are discussed in the Applicable Safety Analyses section of LCO Bases B 3.5.2, ECCS Trains - Operating ($T_{avg} \geq 350°F$) and; B 3.6.3, Containment Spray System.

During a SLB analysis, the delay associated with the interlock between the VCT and RWST isolation valves has been considered and the results show that the DNB design basis is met. The delay, or response time, has been established as [27] seconds with offsite power available and [42] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves and a [10] second stroke time for the VCT valves.

For a large break LOCA analysis, the lower water volume and lower boron concentration limit is used to compute the post-LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since, in the safety analysis, the control rods are all assumed to be out of the core.

The upper limit on boron concentration is used in determining the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of this change in the SI pump suction source is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed equal to the RWST lower temperature limit. If the lower temperature limit is violated, the containment spray further reduces containment pressure which decreases steam venting flow out the break and increases peak clad temperature. The upper temperature limit is used in the small break LOCA analysis and containment integrity analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following a SLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST temperature, boron concentration and volume are process variables that are initial conditions of a Design Basis Accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such they satisfy the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1).

In addition, the RWST is a component that is part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

This LCO establishes the accident analysis requirements for contained volume, boron concentration and temperature of the RWST inventory (Ref. 2). This ensures an adequate supply of borated water is available to cool the core and depressurize the containment in the event of a LOCA or SLB, cool and cover the core in the event of a LOCA, ensure the reactor remains subcritical following a LOCA or SLB, and ensure adequate level exists in the containment sump to support ECCS and CSS pump operation in the recirculation mode.

An additional requirement which affects the required RWST volume is the ability to support continued ECCS and CSS pump operation after the switchover to recirculation. When ECCS and CSS pump suction is transferred to the sump there must be sufficient water in the sump to ensure adequate NPSH for the pumps. The RWST capacity must be sufficient to supply this amount of water without consideration of the inventory added from the accumulators or RCS while accounting for loss of inventory to containment subcompartments and reservoirs due to containment spray operation and areas outside containment due to leakage from ECCS injection and recirculation equipment.

(continued)

BASES (continued)

LCOs (continued)

The limit established for minimum boron concentration ensures that with a minimum RWST level, following a LOCA, the reactor will remain subcritical in the cold condition following mixing of the RWST and RCS water volumes with all control rods removed. The most limiting case would result at end of life with RCS boron concentration at 0 ppm.

The combined limits on water volume and boron concentration of the RWST also ensure a pH value of between [8.0] and [8.2] for the solution recirculated within containment after a LOCA. The pH band minimizes the evolution of iodine and hydrogen and the effects of chloride and caustic stress corrosion on mechanical systems and components in containment.

APPLICABILITY

The RWST OPERABILITY requirements are dictated by the ECCS and CSS OPERABILITY requirements. Since both the ECCS and CSS must be OPERABLE in MODES 1, 2, 3 and 4, the RWST must also be OPERABLE to support their operation.

ACTIONS

A.1

Insufficient water inventory in the RWST could result in insufficient cooling capacity of the ECCS. The 1 hour Completion Time to restore the proper water volume ensures prompt action is taken to restore the RWST to OPERABLE status and is a reasonable time based on operating experience.

B.1

Improper boron concentration could result in a loss of shutdown capability or excessive boric acid precipatation in the core following a LOCA. Water temperature outside the limits could effect containment pressure and peak clad temperatures. The 8 hour Completion Time is reasonable based on industry operating experience and the time required to restore these limits and return the RWST to OPERABLE status.

(continued)

BASES (continued)

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times of Condition A or B are not met, the unit must be placed in MODE 3 within 6 hours and MODE 5 within the following 30. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is reasonable based on industry operating experience, normal cooldown rates and does not challenge safety systems or operators. Continuing the unit shutdown begun in Required Action C.1, an additional 30 hours is a reasonable time based on industry operating experience and normal cooldown rates to reach MODE 5, where this LCO is no longer applicable, from MODE 3 without challenging unit systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

This surveillance ensures the RWST is within the specified temperature assumed in the safety analyses (Ref. 3). The 24 hour frequency is short enough to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through industry operating experience.

SR 3.5.4.2

This surveillance ensures the RWST deliverable volume is ≥ the specified limit. If the level is too low, the RWST would not provide a sufficient initial supply for injection or enough to support continued ECCS and CSS pump operation on recirculation.

Since the RWST volume is normally stable and provided with level alarms, a one week frequency has been established, based on engineering judgment, and has been proven to be acceptable through industry operating experience.

SR 3.5.4.3

This surveillance ensures the boron concentration of the RWST is within the required band. Maintaining the concentration within this band ensures the reactor remains subcritical following a LOCA and that the resulting sump pH is between [8.0] and [8.2]. It also ensures that boron

(continued)

BASES (continued)

SURVEILLANCE REQURIEMENTS (continued)

SR 3.5.4.3

precipitation in the core will not happen and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a one week sampling frequency has been established, based on engineering judgment, and has proven to be acceptable through industry operating experience.

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

2. Title 10 Code of Federal Regulations, Part 50.46, "Acceptance criteria for Emergency Core Cooling Systems for light water nuclear power plants", 1974.

3. WATTS BAR FSAR Chapters 6 and 15.

B 3.5 EMERGENCY CORE COOLING SYSTEM (ECCS)

B 3.5.5 <u>Seal Injection Flow</u>

BASES

BACKGROUND

The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on Reactor Coolant Pump (RCP) seal injection flow limits the amount of ECCS flow which would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions which are required because RCP seal injection flow is not isolated during Safety Injection (SI).

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break Loss Of Coolant Accident (LOCA) event at full power (Ref. 1). This event establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and Steam Line Break (SLB) events also credit the centrifugal charging pumps, but are not limiting in its design.

This LCO ensures that seal injection flow will be limited such that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize core uncovering for a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pump delivers sufficient fluid to maintain RCS inventory.
Seal injection flow is a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

(continued)

BASES (continued)

LCOs

The intent of the LCO limit on seal injection flow is to ensure that flow through the RCP seal water injection line is not greater than that assumed in the safety analysis. This ensures that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points (Ref. 3).

The LCO is not strictly a flow limit but rather a flow limit based on a flow line resistance. The [40] gpm limit is equivalent to [78] gpm for actual accident conditions. In order to establish the proper flow line resistance a pressure and flow must be known. The flow line resistance is determined by assuming the RCS pressure is at normal operating pressure and the centrifugal charging pump discharge pressure is greater than the value specified in this LCO. The centrifugal charging pump discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure would result in more flow being diverted to the RCP seal injection line than at normal operating pressure. The valve settings established at the prescribed centrifugal charging pump discharge header pressure results in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the control valve being fully open, is required since the valve is designed to fail open for the accident condition. With the discharge pressure and control valve position as specified by the LCO a flow limit is established. It is this flow limit that is used in the accident analyses.

APPLICABILITY

The seal injection flow limit is dictated by ECCS flow requirements which include MODES 1, 2, 3, and 4. Therefore RCP seal injection flow must be limited in MODES 1, 2 and 3 to ensure adequate ECCS performance. However the seal injection flow limit is not applicable for MODE 4 because high seal injection flow is less critical due to lower initial RCS pressure and decay heat removal requirements in MODE 4.

(continued)

BASES (continued)

ACTIONS

A.1. and A.2

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this condition, actions must be taken to restore the seal injection flow to below its limit, or to place the unit in a MODE in which this system is not required. The 1 hour Completion Time to restore the seal injection flow is for immediate action that will reduce the flow to within its limit. The adjustment to the flow can be made by either the manual valves or the pressurizer level control valve. If the adjustment is made with the pressurizer level control valve, the operator has four hours from the time the flow is known to be above the limit, to correctly position the manual valves to meet the LCO and thus be in compliance with the accident analysis. The Completion Time minimizes the time exposure of the unit to a LOCA and ensures that seal injection flow is either restored to below its limit or the unit is promptly placed in a MODE in which seal injection flow is not critical. These times are conservative with regards to the Completion Times of LCO 3.5.2, ECCS Trains - Operating ($T_{avg} \geq$ (350°F).

B.1

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown should be commenced. Twelve hours is a reasonable time, based on operating experience, to reach MODE 4 without challenging unit systems or operators. Once MODE 4 is reached, this LCO is no longer applicable.

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

The SR ensures that proper manual seal injection throttle valve position is maintained and hence, proper seal injection flow. The frequency of 31 days has been established, based on engineering judgment, and has proven to be acceptable through operating experience. Because this LCO can only be established by the SR under specific plant conditions, an exemption is permitted to allow the system to stabilize and establish the initial conditions assumed in the accident analysis before the surveillance is performed.

(continued)

BASES (continued)

REFERENCES

1. WATTS BAR FSAR Chapters [6] and [15].

2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

3. Title 10 Code of Federal Regulations, Part 50.46, "Acceptance criteria for Emergency Core Cooling Systems for light water nuclear power plants", 1974.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a free-standing steel vessel, reinforced concrete shell with an annular region separating the two. The containment structure is designed to isolate and contain radioactive material which may be released from the reactor primary system in the event of a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 (Ref. 1) or NRC staff-approved licensing basis (e.g. specified fraction of 10 CFR 100, GDC 19). Additionally, this structure provides shielding from the radioactivity which may be present in the containment atmosphere during normal operations and under post-accident conditions.

The containment is OPERABLE when:

a. Penetrations required to be isolated during accident conditions are either:

   1. Capable of being isolated by an OPERABLE containment automatic isolation system, or

   2. Isolated by locked manual valves, blind flanges, deactivated automatic valves secured in their closed positions, or a closed system, or

   3. Opened under administrative control on selected penetrations;

b. All equipment hatches are closed and sealed;

c. Each air lock is in compliance with the requirements of Technical Specification 3.6.2;

d. Appendix J containment leakage rates are within their required limits; and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

(continued)

BASES (continued)

BACKGROUND (continued)

Containment OPERABILITY requires the existence of a structurally sound containment such that the containment will limit the leakage of fission product radioactivity from containment to the environment under the pressure and temperature conditions that may exist during a Design Basis Accident (DBA).

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, such that, in conjunction with the other containment systems and Engineered Safety Feature Systems, the release of fission product radioactivity subsequent to a DBA will not result in doses in excess of the guideline values specified in 10 CFR 100.

The DBAs which result in a challenge to containment from high pressures and temperatures are a LOCA, a Steam Line Break (SLB), and a Rod Ejection Accident (REA). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or a REA. In the DBA analyses it is assumed that containment is intact at event initiation, such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage (Ref. 2). The containment was designed with an allowable leakage rate of [0.25] % of containment volume per day. This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 3), as L_a; the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting DBA [LOCA]. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leak rate testing. For this unit, L_a = [0.25% by weight] and P_a = [15 psig].

Satisfactory leak rate test results are a requirement for the establishment of containment. The acceptance criteria applied to accidental releases of fission product radioactivity to the environment are given in terms of total radiation dose received by: 1) a member of the general public who remains at the exclusion area boundary for two hours following onset of the postulated fission product

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

radioactivity release, or 2) a member of the general public who remains at the low population zone for the duration of the accident. The limits established in 10 CFR 100 (Ref. 1) are a whole body dose of 25 Rem, or a 300 Rem dose to the thyroid from iodine exposure. The worst case two-hour dose anticipated at the exclusion area boundary occurs following the postulated worst case DBA. The worst case DBA is an overly conservative analysis of the LOCA event for which a significant instantaneous release of fission product radioactivity from the core is postulated.

Containment OPERABILITY involves structures, systems and components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

LCO

Containment OPERABILITY requires the existence of a structurally sound, leak-tight free-standing steel containment.

The maximum allowable containment leak rate (La) at the calculated maximum peak containment internal pressure (Pa) following the design basis LOCA was specified as [0.25%] by the weight of containment air per 24 hours at Pa ([15] psig)(Ref 2). To ensure that this value is not exceeded, containment OPERABILITY must be maintained.

This LCO requires that containment OPERABILITY be maintained. Compliance with this LCO will ensure a containment configuration that will limit leakage to rates assumed in the safety analysis. As a result, radiation exposures at the site boundary will be maintained within regulatory limits 10 CFR 100 (Ref. 1) or some fraction as established in an NRC staff-approved licensing basis following the most limiting DBA.

(continued)

BASES (continued)

LCO (continued)

The provisions of this LCO as delineated in the definition of containment OPERABILITY are implemented as follows:

1. Containment leakage rate requirements are contained in 10 CFR 50, Appendix J and Design Features (Specification 4.2.3). The requirements are implemented to ensure the containment, penetrations and isolation valves do not exceed specified leak rates.

2. Maintaining each containment air lock in accordance with the LCO or ACTIONS of LCO 3.6.2 assures that at least one door in each air lock will be closed and each door will be OPERABLE to assure containment OPERABILITY is maintained.

3. Maintaining each penetration in accordance with the LCO or ACTIONS of LCO 3.6.6 assures that the penetration is capable of being isolated by an OPERABLE containment isolation system or compensatory measures, including administrative control over remote manual valves, have been taken to isolate the penetration to assure containment is maintained.

4. Maintaining each purge valve penetration in accordance with the LCO or ACTIONS of LCO 3.6.6 assures that the penetrations are isolated or compensatory measures have been taken to isolate the penetration to assure containment is maintained.

The measures implemented to meet the above requirements provide assurance that the containment will perform its designed safety function. This function is to mitigate the exposure of the public to the consequences of accidents.

(continued)

BASES (continued)

APPLICABILITY

Maintenance of an OPERABLE containment prevents leakage of radioactive material from the containment. An inoperable containment could cause site boundary doses resulting from a DBA to exceed 10 CFR 100 values or NRC staff-approved safety analysis values.

In MODES 1, 2, 3, and 4, a [MSLB, LOCA, or REA] could cause an increase in radioactive material in containment requiring containment OPERABILITY to be maintained. However, the probability and consequences of these events in MODES 5 and 6 are low because of the reactor coolant system (RCS) pressure and temperature limitations in these MODES. Containment OPERABILITY is, therefore, not required in MODE 5. In MODE 6, fuel handling evolutions are conducted. The requirements for containment during MODE 6 refueling operations are addressed in LCO 3.9.4.

ACTIONS

A.1

In the event the containment is not OPERABLE, it must be restored within one hour. The one-hour Completion Time is based on industry-accepted practice and the low probability of an accident occurring during the one hour when primary containment is not OPERABLE. This period of time to correct a problem is commensurate with the importance of maintaining an OPERABLE containment during MODES 1, 2, 3, and 4.

B.1 and B.2

The plant must be placed in a MODE in which the requirement does not apply if the containment is not OPERABLE in the required time period. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in the following 30 hours. The 6 hours allotted to reach MODE 3 is a reasonable time based on industry operating experience to reach MODE 3 from full power without challenging safety systems. Similarly, the 36 hours allotted is a reasonable time to cooldown to MODE 5 considering that a plant can easily cooldown in such a time frame on one safety system train. In MODE 5, since the probability and consequences of [MSLB, LOCA, or REA] in this MODE are low due to the RCS temperature and pressure limitations.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining an OPERABLE containment requires compliance with the leak test requirements of 10 CFR 50, Appendix J as detailed in the Containment Leak Rate Test Program in Specification 5.9.12. Therefore, this surveillance reflects the leak rate testing requirements with regard to the containment integrated (Type A) and local leak rate (Types B and C) test programs. Specific Appendix J requirements addressed in the Type A, B, and C testing must be performed in accordance with 10 CFR 50, Appendix J as modified by NRC-approved exemptions to Appendix J. These periodic testing requirements verify that the containment leak rate does not exceed the leak rate assumed in the accident analysis. The surveillance frequency is required by Appendix J, and, as such, SR 3.0.2 (which allows surveillance frequency extensions) does not apply. If the sum of local leak rates (Types B and C) exceeds 0.6La, ACTION may be taken in accordance with Technical Specification 3.6.6 (Containment Isolation Valves) to isolate the appropriate penetration(s) if the leaks cannot be repaired within the 4-hour Completion Time. The requirements for the OPERABILITY of the air lock door seals is covered by Technical Specification 3.6.2.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance", 1976.

2. Watts Bar FSAR Section [6.2].

3. Title 10 Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors", 1986.

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, " United States Nuclear Regulatory Commission, February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND	Two containment air locks, which are part of the containment pressure boundary, provide a means for personnel access during all MODES of unit operation. The air locks doors have been designed and certified capable of withstanding a pressure in excess of the maximum peak pressure resulting from the limiting DBA, such that closure of a single door assures containment integrity. Each of the doors is provided with double gasket seals to provide pressure integrity. An air lock door interlock mechanism prevents simultaneous opening of both doors and, therefore, ensures containment OPERABILITY is maintained throughout periods of access to containment. During periods of unit shutdown when containment OPERABILITY 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. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).
APPLICABLE SAFETY ANALYSIS	The safety design basis for the containment is that the containment must withstand the pressure and temperatures of the limiting DBA without exceeding the design leakage rate, such that, in conjunction with the other containment systems and Engineered Safety Feature Systems, the release of fission product radioactivity subsequent to a DBA will not result in doses in excess of the guideline values specified in 10 CFR 100. The DBAs which result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a Steam Line Break (SLB), and a Rod Ejection Accident (REA). In the DBA analyses it is assumed that containment OPERABILITY is intact at event initiation, such that, for the DBAs involving release of fission product radioactivity,

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	release to the environment is controlled by the rate of containment leakage. The containment air locks were designed with an allowable leakage rate of ≤ [0.05La]. Satisfactory leak rate test results are a requirement for the establishment of containment OPERABILITY. The acceptance criteria applied to accidental releases of fission product radioactivity to the environment are given in terms of total radiation dose received by: 1) a member of the general public who remains at the exclusion area boundary for two hours following onset of the postulated fission product radioactivity release, or 2) a member of the general public who remains at the low population zone for the duration of the accident. The limits established in 10 CFR 100 (Ref. 1) are a whole body dose of 25 Rem, or a 300 Rem to the thyroid from iodine exposure. The worst-case DBA is an overly conservative analysis of the [LOCA] event for which an significant instantaneous release of fission product radioactivity from the core is postulated. Containment air lock OPERABILITY involves structures, systems and components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 2).
LCO	Compliance with this LCO ensures a containment configuration that limits containment leakage rates to less than the values assumed in the safety analyses. As a result, radiation exposures at the SITE BOUNDARY will be maintained within the limits of 10 CFR 100 following the limiting DBA.
APPLICABILITY	Containment air lock OPERABILITY must be maintained in MODES 1, 2, 3, and 4, when a DBA could cause a substantial increase in fission product radioactivity in containment. In MODES 5 or 6, containment OPERABILITY is not required because of the pressure and temperature limitations in these MODES: LCO 3.9.4, Containment Building Penetrations, contains requirements for maintaining containment closure during CORE ALTERATIONS or movement of irradiated fuel within containment in MODE 6.

(continued)

BASES (continued)

ACTIONS

A.1, A.2.1, A.2.2.1, AND A.2.2.2

Air locks are provided with two doors, each of which is designed to seal against the maximum containment pressure resulting from the limiting DBA. Should an air lock become inoperable as a result of an inoperable air lock door or an inoperable door interlock, power operation may continue provided that at least one OPERABLE air lock door is closed within 15 minutes. In addition, either the inoperable equipment must be repaired within a Completion Time of 24 hours per Required Action A.2.1 or the OPERABLE air lock door must be closed and locked within the Completion Time of 24 hours per Required Action A.2.2.1, and verified locked closed at least every 31 days per Required Action A.2.2.2.

The 24 hour Completion Times of Required Actions A.2.1 and A.2.2.1 are based on engineering judgment that reasonable time be allowed to perform Required Actions without prolonging the time that containment OPERABILITY is not ensured. The note in Required Actions allows access through a door that is used to satisfy the subject Required Actions. However, this access is only permitted for repair of inoperable air lock equipment and to perform required surveillance testing such as for the ice bed or to make any other necessary containment entries. If the Required Actions cannot be completed in the allotted time, Condition C requires that the unit be placed in a MODE in which the LCO does not apply.

OPERABILITY of air locks is required to ensure that containment is maintained. Required Action A.2.2.2 ensures that at least one air lock door is closed and maintained closed until the air lock is restored to an OPERABLE status.

B.1 and B.2

OPERABILITY of air locks is required to ensure that containment OPERABILITY is maintained. Should an air lock become inoperable for reasons other than stated in Condition A, the air lock leak tight integrity must be restored within 24 hours or actions must be taken to place the unit in a condition for which the LCO does not apply. Required Actions B.1 and ensure that at least one air lock door is closed and maintained closed until the air lock is restored to an OPERABLE status per Required Action B.2.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2 (continued)

The Completion Time of 15 minutes per Required Action B.1 is considered the shortest practical time for an operator to evaluate the situation and take the Required Action . The Completion Time of 24 hours is based on engineering judgment that a reasonable time period should be allowed to restore the air lock to OPERABLE status without unnecessarily prolonging the time that containment OPERABILITY is not ensured. The first note in the Required Action states that if both door seals have catastrophically failed then containment shall be declared inoperable in accordance with LCO 3.6.1, Containment. This is done because the leakage past both seals would most likely invalidate the overall containment leakage limit. The second note in the Required Action allows entry and exit to perform repairs of the affected air lock components, perform required surveillances, and to make necessary containment entries.

C.1 and C.2

The unit must be placed in a MODE in which the LCO does not apply if the respective Required Actions and associated Completion Times cannot be met. This is done be placing the unit in MODE 3 in 6 hours and in MODE 5 within the next 30 hours. The 6 hours allowed to reach MODE 3 is a reasonable time, based on industry operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is reasonable time considering that a unit can easily cooldown in such a time frame on one safety system train.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

The periodic leak rate testing program is performed in accordance with 10 CFR 50, Appendix J as modified by approved exemptions, as contained in the Containment Leak Rate Testing Program. The leak rate tests verify that the actual leakage rates from containment are less than or equal to that assumed in the limiting DBA analysis.

The note states that an inoperable air lock door does not invalidate the previous successful performance of an overall air lock leakage test.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.2

The air lock door 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 peak post-accident containment pressure, closure of either door will maintain containment OPERABILITY. Thus, the door interlock feature ensures that containment OPERABILITY is maintained while the air lock is being used for personnel entry and exit. Periodic testing of this interlock provides assurance that the interlock will function as designed, and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, a frequency of 6 months is considered adequate to detect degradation. As such, the frequency is based on engineering judgment.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area Low Population Time and Population Center Distance", 1976.

2. 52F3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Spray System (CSS)

BASES

BACKGROUND

The CSS is designed to furnish 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 spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to less than the guidelines of 10 CFR 100 (Ref. 1).

The CSS consists of two separate trains of equal capacity, each capable of meeting the system design bases spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, one spray header, nozzles, valves, piping, instrumentation, and controls. Each train is powered from a separate Engineered Safety Feature (ESF) bus. The Refueling Water Storage Tank (RWST) supplies borated water to the CSS during the injection phase of operation. In the recirculation mode of operation, CSS pump suction is transferred from the RWST to the containment recirculation sump.

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the containment spray system heat removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the CSS.

The CSS and RHR System provide a spray of cold or subcooled borated water into the upper regions of containment to limit the containment pressure and temperature during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the CSS during the injection phase. In the recirculation mode of operation heat is removed from the containment sump water by the CSS and RHR heat exchangers. Each train of the CSS, supplemented by a train of RHR spray, provides adequate spray coverage to meet the system design requirements for containment heat removal.

(continued)

BASES (continued)

BACKGROUND (continued)

The CSS is actuated either automatically by a Containment Hi-Hi pressure signal or manually. An automatic actuation opens the CSS pump discharge valves, starts the two CSS pumps and begins the injection phase. A manual actuation of the CSS requires the operator to actuate two separate switches on the main control board to begin the same sequence (manual Phase "B"), or manually start the pumps with the main control room handswitches. The injection phase continues until a RWST Level Lo-Lo alarm is received. The Lo-Lo level alarm for the RWST signals the operator to manually align the system to recirculation mode. To execute the transfer to take suction from the containment sump, the operator must stop the CSS pumps, close the suction valves from the RWST, align cooling water to the CSS heat exchangers then open the suction valves from the containment sump. The CSS in recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the CSS in the recirculation mode is controlled by the operator using the Emergency Operating Procedures.

The RHR spray operation is initiated manually when required by the Emergency Operating Procedures after the Emergency Core Cooling System is operating in the recirculation mode. If the switchover to recirculation occurs prior to one hour after initiation of the LOCA, RHR spray operation may be commenced one hour after initiation of the LOCA. If switchover to recirculation occurs later than one hour after initiation of the LOCA, RHR spray operation may be commenced after completion of the switchover procedure. The reason for the one hour minimum is to ensure adequate RHR flow to the core to remove the initial decay heat.

The CSS is a containment ESF System. It is designed to ensure that the heat removal capability required during the post-accident period can be attained. The operation of the CSS, together with the ice condenser, is more than adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post-blowdown period, the Air Return Fan System (ARFS) is automatically started. The ARFS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

(continued)

BASES (continued)

BACKGROUND (continued)

After the Emergency Core Cooling System (ECCS) is aligned to the recirculation mode, the RHR sprays are available to supplement the CSS, if required, in limiting containment pressure. The operation of these systems provide the required heat removal capability to limit post-accident conditions to less than the containment design values in accordance 10 CFR 50, Appendix A, General Design Criteria (GDC) 38 (Ref. 2).

The CSS protects the OPERABILITY of the containment by limiting the temperature and pressure that could be expected following a DBA. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CSS protects the integrity of containment by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to containment are the Loss of Coolant Accident (LOCA) and the Steam Line Break Accident (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, [in regards to containment ESF systems, assuming the loss of one ESF bus, resulting in one train of CSS, RHR, and ARS inoperable] (Ref. 3).

The DBA analyses show that the maximum peak containment pressure results from the [LOCA] analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the [SLB] analysis and [was calculated to be less than the containment design temperature.]

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The inadvertent actuation of the CSS is precluded by the design of the ESF Actuation System and the main control board switches. Inadvertent actuation of the CSS is evaluated in the analysis and the resultant reduction in containment pressure is calculated. The maximum calculated reduction in containment pressure resulted in a containment external pressure load below the containment design external pressure load.

The modeled CSS actuation from the containment OPERABILITY analysis is based upon a response time associated with exceeding the containment pressure Hi-Hi signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses peak calculated containment temperature and pressure responses. [The CSS total response time of [/] seconds, (with/without offsite power) is composed of the following: signal delay [ ] seconds, diesel generator startup [ ] seconds (for loss of offsite power), and system startup time [ ] seconds.] Surveillance Requirement [ ] of LCO 3.3.2, ESF Actuation System, addresses the response time testing requirements.

The CSS is a system that is part of the primary success path and which actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 5).

LCO

The CSS trains are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of CSS, ARFS, RHR sprays and the ice bed are required to provide the heat removal capability assumed in the safety analyses for an OPERABLE containment.

(continued)

BASES (continued)

LCO (continued)

To ensure one train of CSS is available, two trains of the CSS must be OPERABLE. This will ensure that at least one train of CSS will operate assuming the worst case single failure occur in the ESF power supply.

APPLICABILITY

LCO 3.6.3 requires the CSS OPERABLE in MODES 1, 2, 3, and 4 to protect the integrity of the containment by limiting the temperature, pressure and iodine fission product release that could be experienced following a DBA.

In MODES 1, 2, 3, and 4 a DBA could cause an increase in containment pressure and temperature requiring the operation of the CSS. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, the CSS is not required to be OPERABLE in these MODES to protect containment.

ACTIONS

A.1

With one of the required trains of CSS inoperable, the inoperable spray train must be restored to OPERABLE status within the Completion Time of 72 hours before action must be taken to reduce power. The specified time period is consistent with other LCOs for the loss of one train of an ESF system, and is a reasonable repair time for many repairs. As such, the Completion Time of 72 hours is based on engineering judgment.

B.1 and B.2

If the inoperable train of CSS cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours. While in MODE 3, an additional 48 hours is allowed to restore the single inoperable train of CSS to OPERABLE status. The probability and consequences of a LOCA or SLB in MODE 3 is less than in MODES 1 and 2. If the inoperable train of CSS cannot be restored in 48 hours the unit must be placed in MODE 5 in the next 30 hours giving a total time of 84 hours to be in MODE 5.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2 (continued)

The 6 hours allowed to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5, the CSS is no longer required OPERABLE to protect the integrity of containment due to the pressure and temperature limitations of this MODE.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Verifying the correct alignment of manual, power-operated, and automatic valves, excluding check valves, in the CSS provides assurance that the proper flowpath exists for CSS operation. This surveillance requirement does not apply to valves which are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. The frequency of 31 days was established based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.3.2

Periodic surveillance testing of centrifugal pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 6). This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies that both 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. The acceptable limits for the pump's performance are contained in the Inservice Inspection and Testing Program. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements but allowable pump degradation is based on FSAR accident analysis assumptions, [not] ASME Section XI. No additional requirements are specified.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

In order for the CSS to maintain an OPERABLE containment following a DBA, automatic valves, excluding check valves, must position on a CSS actuation signal. These surveillances ensure each automatic valve actuates on receipt of a Containment Spray actuation signal. The frequency of 18 months is typical of other technical specification valve and pump start frequencies. As such, the frequency of 18 months is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.3.4

In order for the CSS to maintain an OPERABLE containment, the CSS pumps must start on a CSS actual or simulated containment pressure Hi-Hi signal. The frequency of 18 months is typical of other technical specification valve and pump start frequencies. Operational experience at other facilities has demonstrated that 18 months is acceptable.

SR 3.6.3.5

The performance of this surveillance verifies each spray nozzle is unobstructed and spray coverage of the containment will meet its design bases objective. An air, smoke or equivalent test is performed. Due to the passive design of the spray header and its normally dry state, a frequency of 5 years is considered adequate to detect performance degradation. As such, the frequency was based on engineering judgment, and has been shown to be acceptable through industry operating experience.

These surveillances meet the requirements of GDC 40 (Ref.2).

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 100.11, "Determination of exclusion area, low population zone, and population center distance".

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants,"

   a. No. 38, "Containment heat removal"

   b. No. 40, "Testing of containment heat removal systems"

3. Watts Bar FSAR, Section [6.5.2].

4. Title 10 Code of Federal Regulations, Part 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants".

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements For Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, February 6, 1987.

6. <u>ASME Boiler and Pressure Vessel Code</u>, Revision [ ], Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", American Society of Mechanical Engineers, New York, [1983].

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Air Return Fan System (ARFS)

BASES

BACKGROUND

The ARFS is designed to assure the rapid return of air from the upper to the lower containment compartments after the initial blowdown following a Design Basis Accident (DBA). This recirculation of containment air back into lower containment and subsequently back up through the ice condenser assists in cooling the containment atmosphere and limiting the post-accident pressure and temperature in containment to less than the design values. The reduction of containment pressure and temperature limits the release of fission product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1).

The ARFS draws air from the dome of the containment vessel, from the reactor cavities, and from the ten dead-ended (pocketed) spaces in the containment where there is potential for the accumulation of hydrogen. The ten dead-ended spaces are the four steam generator enclosures, the pressurizer enclosure, the four accumulator spaces and the instrument room. Each fan will mix air from the enclosed areas in the lower volume to the general lower volume atmosphere to prevent excessive localized hydrogen build-up following a DBA. The air flow is sufficient to limit the local concentration of hydrogen to less than [3%] when the containment average concentration is [2%].

The ARFS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper and hydrogen collection headers. Each train is powered from a separate Engineered Safety Feature (ESF) bus.

The ARFS fans are automatically started by the Containment Pressure Hi-Hi signal [10 $\pm$ 1] minute after the containment pressure reaches [2.81] psid . The fans displace air from the upper compartment to the lower compartment, returning air which was displaced by the high energy line break blowdown to the lower compartment, and equalizing pressures throughout containment.

(continued)

BASES (continued)

BACKGROUND (continued)

After discharge into the lower compartment, air flows with steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow is condensed. The air flow returns to the upper compartment through the top deck doors in the upper portion of the ice condenser compartment. The ARFS fans operate continuously after actuation, circulating air through the containment volume and taking suction from potential hydrogen pockets in containment.

The ARFS is a containment ESF System. It is designed to ensure the heat removal capability required during the post-accident period can be attained. The operation of the ARFS in conjunction with the ice bed, the Containment Spray System (CSS) and Residual Heat Removal (RHR) spray provide the required heat removal capability to limit post-accident conditions to less than the containment design values in accordance 10 CFR 50, Appendix A, General Design Criteria (GDC) 38 (Ref. 2).

The ARFS protects the integrity of the containment by limiting the temperature and pressure that could be expected following a DBA. Maintaining an OPERABLE containment limits leakage of fission product radioactivity from containment to the environment. An inoperable containment could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ARFS protects the integrity of containment by limiting the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered relative to an OPERABLE containment are the Loss of Coolant Accident (LOCA) and the Steam Line Break Accident (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively. The postulate DBAs are analyzed, [in regards to containment ESF system, assuming the loss of one ESF bus, resulting in one train of CSS, RHR, EGTS, and ARFS inoperable] (Ref. 3).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The DBA analyses show that the maximum peak containment pressure results from the [LOCA] analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the [SLB] analysis and [was calculated to be less than or equal to the containment design temperature.]

[The ARFS total response time of [/] seconds, (with/without offsite power) is composed of the following: signal delay [ ] seconds, diesel generator startup [ ] seconds (for Loss of Offsite Power), and ARFS startup time of [ ] seconds.] Surveillance Requirement [ ] of LCO 3.3.2, ESF Actuation System addresses the response time testing requirements.

The ARS is a system that is part of the primary success path and which actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 5).

(continued)

BASES (continued)

LCO

The ARFS trains are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of the ARFS with hydrogen collection ducts is required to provide the minimum air recirculation for heat removal and hydrogen mixing assumed in the safety analyses for containment OPERABILITY. To ensure this requirement is met, two trains of the ARFS with hydrogen collection ducts must be OPERABLE. This will ensure that at least one train will operate assuming the worst case single failure occurs in the ESF power supply.

APPLICABILITY

LCO 3.6.4 requires the ARFS OPERABLE to protect the integrity of the containment by limiting the temperature and pressure that could be experienced following a DBA and to provide post-accident hydrogen mixing. The ARFS together with the CSS, RHR Spray, and Ice Bed provide the required heat removal capability to limit pressure and temperature. In MODES 1, 2, 3, and 4 a DBA could cause an increase in containment pressure and temperature requiring the operation of the ARFS. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, the ARFS is not required to be OPERABLE in these MODES to protect containment OPERABILITY.

ACTIONS

A.1

With one of the required trains of ARFS inoperable, the inoperable ARFS train must be restored to OPERABLE status within the Completion Time of 72 hours before action must be taken to reduce power. The specified time period is consistent with other LCOs for the loss of one train of an ESF system, and is a reasonable repair time for many repairs. As such, the Completion Time of 72 hours is based on engineering judgment.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If the inoperable train of ARFS cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is reasonable, considering that a unit can easily cooldown in such a time frame on one safety system train. In Mode 5, the ARFS is no longer required OPERABLE to protect the integrity of containment due to the pressure and temperature limitations of this MODE.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Demonstrating each ARS fan starts on a simulated Phase B signal and operates for 15 minutes is sufficient to ensure that all fans of both trains are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The frequency of 18 months was based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.4.2

Demonstrating fan motor current at rated speed with the return air dampers closed confirms one operating condition of the fan. This test is indicative of overall fan/motor performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The frequency of 92 days conforms with the testing requirements for similar ESF equipment. As such, the frequency was based on engineering judgment, and has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3

Verifying the OPERABILITY of the return air damper provides assurance that the proper flowpath will exist when the fan is started. By applying the correct counterweight the damper operation can be confirmed. The frequency of 92 days is based on engineering judgment. It takes into account the importance of the dampers, their location, physical environment, and probability of failure.

These surveillances meet the requirements of GDC 40 (Ref. 2).

REFERENCES

1. Title 10 Code of Federal Regulations, Part 100.11, "Determination of exclusion area, low population zone, and population center distance".

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants,"

   a. No. 38, "Containment heat removal"

   b. No. 40, "Testing of containment heat removal systems"

3. Watts Bar FSAR, Section [6.8].

4. Title 10 Code of Federal Regulations, Part 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants".

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements For Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Emergency Gas Treatment System (EGTS)

BASES

BACKGROUND

The EGTS ensures that radioactive materials leaking from the containment atmosphere into the Shield Building following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential release of radioactive material, principally iodine, to within values specified in 10 CFR 100, paragraph 100.11. (Ref. 1).

The EGTS consists of two separate and redundant trains. Each train includes a heater, prefilter, demister, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves and/or dampers and instrumentation also form part of the system. The demisters and heaters function to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates filtered ventilation of the Shield Building following receipt of a Phase "A" isolation signal from either unit. The system design complies with the 10 CFR 50, Appendix A, General Design Criteria (GDC) 41, 42, and 43 requirements for containment cleanup systems (Ref. 2) and is described in the FSAR (Ref. 3).

The demister is included for moisture (free water) removal from the gas stream. The heaters are used to heat the gas stream which lowers the relative humidity. Continuous operation of each train for at least 10 hours per month with the heaters on reduces moisture buildup on (drys) the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.

The Containment annulus vacuum fans maintains the annulus at [5] inches water gauge vacuum during normal operations. The initial annulus vacuum is an initial condition in the safety analysis and, therefore, must be maintained within limits.

(continued)

BASES (continued)

BACKGROUND (continued)	The EGTS reduces the radioactive content in the Shield Building following a DBA. Loss of the EGTS could cause SITE BOUNDARY doses, in the event of a DBA, to exceed the values given in 10 CFR 100 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The EGTS design basis is established by the consequences of the limiting DBA which is a Loss of Coolant Accident (LOCA). The accident analysis (Ref. 4) assumes that only one train of the EGTS is functional due to a single failure which disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA. The EGTS is part of the primary success path which functions or actuates to mitigate a Design Bases Accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5).
LCO	The EGTS trains each have separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of the EGTS is required to provide the minimum fission product removal assumed in the safety analysis. Two trains of the EGTS must be OPERABLE to ensure that these minimum requirements are met. This will ensure that at least one train will operate, assuming that one train is disabled by a single failure.
APPLICABILITY	LCO 3.6.5 requires the EGTS to be OPERABLE in MODES 1, 2, 3, and 4 because of the potential for a fission product release following a DBA. The design of the EGTS is based on a MODE 1 LOCA. Less severe LOCAs and leaks also require OPERABILITY of the EGTS in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of a DBA are low due to the RCS pressure and temperature limitations of these MODES. As such, the EGTS is not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1

When one train of the EGTS is determined to be inoperable, action is required to restore the system to OPERABLE status. A Completion Time of 7 days is permitted to restore the system to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering that the remaining train can provide all of the required air cleanup capability and is an adequate time to make most repairs.

B.1

Maintaining the annulus pressure at $\geq$ [5] inches water gauge vacuum during normal operation ensures that after a LOCA, the annulus can be maintained at $\geq$ [0.5] inches water gauge vacuum. The Completion Time of 8 hours is based on engineering judgment. The note says that the requirement to maintain the annulus pressure within limits is not applicable during venting operations, required annulus entries, or Auxiliary Building isolations not exceeding 1 hour in duration.

C.1 and C.2

If the EGTS cannot be restored to OPERABLE status or the annulus vacuum pressure restored within limits in the required Completion Time, the unit must be placed in a MODE where a DBA is not credible. This is done by placing the unit in MODE 3 in a Completion Time of 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5 the EGTS is no longer required to be OPERABLE to limit the release of radioactive material; due to the RCS pressure and temperature limitations of the MODE.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

This test provides periodic confirmation of the OPERABILITY of each train of the EGTS. Additionally, continuous operation of the system for 10 hours with the heaters on assures that the filters and charcoal adsorbers are dry for maximum effectiveness. The frequency of 31 days is based on engineering judgment taking into account the importance of the system, the normally mild environment and operating conditions and the low likelihood of failure on demand. Additionally, this frequency has been shown to be acceptable through industry operating experience.

SR 3.6.5.2

The requirements and acceptance criteria of the filter testing is done in accordance with the Ventilation Filter Testing Program. This program is outlined in Specification 5.9.13.

SR 3.6.5.3

The automatic startup test verifies that both trains of equipment start on receipt of a containment Phase "A" test signal. The frequency of 18 months is considered to be adequate considering the reliability of electronic actuation systems. Additionally, this frequency has been shown to be acceptable through industry operating experience.

SR 3.6.5.4

To ensure that each damper actuates to its correct position, the dampers are checked every 18 months. The frequency is based on engineering judgement.

SR 3.6.5.5

The proper function of the fans, dampers, filters, adsorbers, etc. as a system is verified by the ability to produce the required negative pressure ($\geq$ [0.5] inch water gage) relative to the Mechanical Equipment Room El. 772 during test operation within [ ] minute with an inleakage of $\leq$ [500] cfm. The negative pressure assures that the building is adequately sealed and that leakage from the building will be prevented, since outside air will be drawn in by the low pressure. The negative pressure must be established within the time limit to assure that no significant quantity of radioactive materials leak from the

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.5 (continued)

shield building prior to developing the negative pressure. The frequency of 18 months is consistent with the Regulatory Guide 1.52, (Ref. 7) guidance for functional testing.

SR 3.6.5.6

The proper pre-accident conditions in the annulus of ≥ [5] inches water gauge vacuum is verified every 24 hours. The frequency is based on engineering judgement.

REFERENCES

1. Title 10, Code of Federal Regulations , Part 100, 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".
2. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants;
   GDC 41, "Containment atmosphere cleanup",
   GDC 42, "Inspection of containment atmosphere cleanup systems", and
   GDC 43, "Testing of containment atmosphere cleanup systems".
3. Watts Bar FSAR Section [6.5], "Fission Product Removal and Control Systems".
4. Watts Bar FSAR, Chapter [15], "Accident Analysis".
5. NRC Interim Policy Statement, 52FR3788, "Technical Specification Improvements For Nuclear Power Reactors", February 6, 1987.
6. ANSI/ASME N510-[1975], "Testing of Nuclear Air Cleaning Systems".
7. Regulatory Guide 1.52, Rev. [2], "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants".
8. NRC Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications of ESF Cleanup Systems", March 2, 1983.
9. ANSI/ASME N509-[1976], "Nuclear Power Plant Air Cleaning Units and Components".

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Isolation Valve

BASES

BACKGROUND

In order to minimize containment leakage, penetrations not required for accident mitigation during a DBA are provided with isolation devices. These isolation devices consist of either passive devices or active automatic devices. Locked closed manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the flow path secured), blind flanges, and closed systems are considered passive devices. Closed systems are those systems designed in accordance with General Design Criterion 57 (Ref. 1) and are identified as such in FSAR, Section 6.2.4 (Ref. 2). Check valves, or other automatic valves designed to close following an accident without operator action, are considered active devices. Two isolation devices are provided in series 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.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a Safety Injection Signal. The Phase "A" Isolation Signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a Containment Pressure--Hi-Hi Signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals above, the purge and exhaust valves receive an isolation signal on [Containment Radiation--High] and [purge exhaust radiation-High]. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a DBA.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment isolation systems provide the means of isolating fluid systems that pass through containment penetrations so as to confine to the containment any radioactivity that may be released in the containment following a design basis event. The containment isolation systems are required to function following a design basis event to isolate nonsafety-related fluid system penetrating the containment. This limits the release of fission product inventory subsequent to a DBA such that the doses in excess of the guideline values in 10 CFR 100 are not exceeded.

LCOs

Compliance with this LCO ensures a containment configuration that limits containment leakage rates to less than the values assumed in the safety analyses. As a result, radiation exposure at the SITE BOUNDARY will be maintained within the limits of 10 CFR 100 following the limiting DBA.

APPLICABILITY

Containment isolation valves must be OPERABLE in MODES 1, 2, 3, and 4 when a DBA could cause a substantial increase in fission product inventory in containment. In MODE 5, containment isolation valves are not required to be OPERABLE because of the pressure and temperature limitations, except as necessary to support reduced RCS inventory operations. LCO 3.9.4, Containment Building Penetrations, contains requirements for maintaining containment OPERABLE during CORE ALTERATIONS or movement of irradiated fuel within containment in MODE 6.

ACTIONS

A.1, A.2.1, A.2.2.1, and A.2.2.2

Required Actions A.1, A.2.1, A.2.2.1, and A.2.2.2 address penetrations that are provided with 2 isolation devices. The note in condition A states that this condition is not applicable to those penetrations with only one containment isolation valve and a closed system. Should one or more devices become inoperable on one or more penetrations, the potential exists that allowable leakage rates may be exceeded in a DBA. The line with the inoperable valve must be verified to have at least one isolation valve OPERABLE in

(continued)

BASES (continued)

ACTIONS (continued)

A.1, A.2.1, A.2.2.1, and A.2.2.2 (continued)

each affected open penetration within 1 hour. Preferably, the isolation device or devices will be restored to OPERABLE within 4 hours. Otherwise each affected penetration must be isolated in accordance with A.2.2.1 by use of at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve inside containment with flow through the valve secured. In addition, A.2.2.2 states that each affected penetration must be verified to be isolated once per 31 days for valves not in containment or the annulus, and prior to entering MODE 4 from MODE 5, but not more often than once per 92 days for valves inside containment or the annulus.

B.1, B.2.1, and B.2.2

Required Actions B.1, B.2.1, and B.2.2 are only applicable to those penetrations with only one containment isolation valve and part of a closed system. Seven days are allowed under B.1 and B.2.1 to either restore the valve to OPERABLE or to isolate each affected penetration by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. This time is allowed due to the nature of a closed system which doesn't communicate with the RCS or the containment atmosphere. In addition, B.2.2 requires that each affected penetration be verified to be isolated once per 31 days for valves not in containment or the annulus.

C.1, C.2, and C.2.2

The containment purge valves are provided to have a means of cleaning up the containment atmosphere during normal operations. With one or more containment purge valves not within limits, the leakage must be restored to within limits within 24 hours. If the leakage cannot be restored within 24 hours, each affected penetration must be isolated by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange within 24 hours. These times are consistent with other allowed outage times for containment isolation. In addition, if the purge valves are not within limits, SR 3.6.6.6 must be performed on the valves with resilient seals.

(continued)

BASES (continued)

ACTIONS (continued)

D.1 and D.2

The unit must be placed in a MODE in which the LCO does not apply if the respective Required Actions for a particular Condition cannot be completed within their required Completion Times. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 within the next 30 hours. The 6 hours allowed to reach MODE is a reasonable time, based on industry operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is reasonable, considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5 containment OPERABILITY is not required due to the RCS pressure and temperature limitations of MODE 5 unless required for reduced RCS inventory conditions.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

The 24-inch containment lower compartment purge supply and/or exhaust isolation valves must be verified to be physically restricted to $\leq$ [50] degrees open every 31 days. This is to ensure that the proper closure times can be met upon receipt of an isolation signal. The 31 day frequency is based on engineering judgement.

SR 3.6.6.2

Containment isolation manual valves, blind flanges and deactivated automatic valves which are located outside containment, the annulus or the North or South Valve Vault Rooms which are required to be closed during accident conditions must be verified closed every 31 days.

Note 1 states that valves and blind flanges in high radiation, or hazardous areas may be verified by the use of administrative controls. Note 2 allows valves to be opened intermittently under administrative controls. Note 3 states that this SR is not required to be met while those valves are under administrative control.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.3

The containment isolation manual valves and blind flanges which are inside the areas excluded by SR 3.6.6.2, (containment, annulus, and the North and South valve vaults) and required to be closed during an accident must be verified closed. This is done prior to entering MODE 4 from MODE 5 but not more often than once per 92 days.

SR 3.6.6.4

This surveillance requires that the isolation time of each power-operated and each automatic containment isolation valve be demonstrated to be within limits in accordance with the Inservice Inspection and Testing Program.

SR 3.6.6.5

This surveillance ensures that each containment isolation valve actuates to its isolation position by demonstrating that it closes after reception of an actual or simulated Phase "A", Phase "B", or Containment Vent Isolation signal. This is performed every 18 months and is considered adequate to ensure proper operation.

SR 3.6.6.6

For containment purge valves with resilient seals, additional required leak rate testing must be performed in accordance with the Containment Leak Rate Testing Program every 184 days and within 92 days after opening the valve. The note states that the test results shall be evaluated against the acceptance criteria of SR 3.6.1.1, Containment Operability. This ensures that leakage is within the assumptions used for containment OPERABILITY.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants ", No. 57 " Closed System Isolation Valves", 1978.

2. Watts Bar FSAR Section 6.2.4, Containment Isolation Valves.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Containment Internal Pressure

BASES

BACKGROUND

The containment internal pressure is a process variable which is monitored and controlled. The containment internal pressure limits are derived from the input conditions used in the containment Design Basis Accident (DBA) analyses. Limiting the containment internal pressure and air temperature, limits the pressure that could be expected following a DBA, thus protecting the integrity of containment. Maintaining an OPERABLE containment limits leakage of fission product radioactivity from containment to the environment. An inoperable containment could cause SITE BOUNDARY doses, in the event of a DBA, to exceed the guidelines given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analysis which establishes the maximum peak containment internal pressure. The limiting DBAs considered relative to an OPERABLE containment are the Loss of Coolant Accident (LOCA) and Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two LOCAs or SLBs are assumed to occur simultaneously, or consecutively. The postulated DBAs are analyzed in regards to containment Engineered Safety Features (ESF) systems, assuming the loss of one ESF bus, resulting in one train of Containment Spray System (CSS) , Residual Heat Removal Spray System, and Air Return Fan System to be inoperable. The containment analyses, References 2 and 3, for the DBA shows that the maximum peak containment pressure, P_a, results from the limiting design basis LOCA.

The maximum design internal pressure for containment is [15] psig. The initial pressure condition used in the containment DBA analyses was [+0.3] psig. This resulted in a maximum peak pressure of [12.3] psig which is less than the maximum design internal pressure for containment.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an external pressure load of [2.0] psid. The inadvertent actuation of the Containment Spray System (CSS), (Ref. 4), is analyzed to determine the reduction in containment pressure. The initial pressure condition used in this analysis was [-0.1] psid. This resulted in a minimum pressure inside containment of [ ] psig. This results in a load across containment of [ ] psid which is less than the design load.

Containment internal pressure is a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 5).

LCOs

The containment internal pressure LCO limits have been established to ensure that containment operation is maintained within the limits assumed in the containment design bases analyses. Satisfying the LCO limits ensures an OPERABLE containment is maintained.

APPLICABILITY

LCO 3.6.7 establishes the containment internal pressure limits for MODES 1, 2, 3, and 4. The occurrence of a DBA LOCA, SLB or inadvertent actuation of the CSS in these MODES could compromise containment OPERABILITY. Operation within these limits ensures an OPERABLE containment is maintained in the event of a DBA.

In MODES 5 and 6 the possibility and consequences of a SLB or LOCA are low due to the RCS pressure and temperature limitations of these MODES. In addition, the CSS is not required to be available in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1

With containment internal pressure outside the subject LCO limits, it must be restored to within the limits within the Completion Time of 1 hour or else action must be taken to reduce power. This action must be taken to return the unit to within the bounds of the containment design bases analyses. The Completion Time of 1 hour is sufficient to correct minor problems or to prepare the unit for an orderly shutdown and entry into Condition B.

B.1 and B.2

The unit must be placed in a MODE in which the LCO does not apply if the containment internal pressure cannot be restored to within its limits within the Completion Time of 1 hour. This is done by placing the unit in at least MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed to reach MODE 3 is a reasonable time based on industry operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is a reasonable time considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5, maintaining containment internal pressure is no longer required.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying that the containment internal pressure is within limits ensures that containment operation remains within the limits assumed in the containment design bases analyses. The frequency of 12 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".

2. Watts Bar FSAR, Section 6.2.

3. Watts Bar FSAR, Chapter 15, Accident Analysis.

4. Watts Bar FSAR, Section 6.2.

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, United States Nuclear Regulatory Commission, February 6, 1987.77

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Containment Air Temperature

BASES

BACKGROUND

Containment air temperature is a process variable which is monitored and controlled. The containment average air temperature limit is derived from the input conditions used in the containment Design Basis Accident (DBA) analyses. Limiting the containment air temperature and internal pressure, limits the temperature and pressure that could be expected following a DBA, thus protecting the integrity of containment. Maintenance of Containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed the guidelines given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses which establishes the maximum peak containment internal pressure and in establishing the environmental qualification operating envelope. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment, References 2 and 3. The limiting DBAs considered relative to containment OPERABILITY are the Loss of Coolant Accident (LOCA) and Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two LOCAs or SLBs are assumed to occur simultaneously, or consecutively. The postulated DBAs are analyzed in regards to containment Engineered Safety Features (ESF) systems, assuming the loss of one ESF bus, resulting in one train of Containment Spray System (CSS), Residual Heat Removal System, and Air Return System to be inoperable.

The limiting DBA for the maximum peak containment air temperature is a SLB. For the upper compartment, the initial containment average air temperature assumed in the design basis analyses (Ref. 2), [110]°F, resulted in a maximum containment air temperature of [ ]°F. The design temperature is [ ]°F. For the lower compartment, the initial average containment air temperature assumed in the design basis analyses, [120]°F, resulted in a maximum containment air temperature of [ ]°F. The design temperature is [ ]°F.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS (continued)

The temperature upper limits are used to establish the environmental qualification operating envelope for both containment compartments. The maximum peak containment air temperature for both containment compartments [was calculated to be less than the containment design temperature.] [was calculated to exceed the containment design temperature for a short period of time during the transient. However, the basis of the containment design temperature is to ensure the OPERABILITY of safety-related equipment inside containment. Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough, that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperatures are acceptable for the DBA SLB].

The temperature upper limits are also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (CSS), (Ref. 4), for both containment compartments.

The containment pressure transient is sensitive to the initial air mass in containment and therefore the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is the LOCA. The temperature lower limits, [85]° F for the upper compartment and [100]°F for the lower compartment, are used in this analyses to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded in either containment compartment.

Containment average air temperature is a process variable that is an initial condition of a DBA or transient analyses that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 5).

LCOs

The containment average air temperature LCO limits for the upper compartment and for the lower compartment have been established to ensure that containment operation is maintained within the limits assumed in the containment design bases analyses. Satisfying the LCO limit ensures containment OPERABILITY is maintained.

(continued)

BASES (continued)

APPLICABILITY

LCO 3.6.8 establishes the containment average air temperature limits for MODES 1, 2, 3, and 4. The occurrence of a DBA LOCA, SLB or inadvertent actuation of the CSS in these MODES could compromise containment OPERABILITY. Operation within these limits ensures containment OPERABILITY is maintained in the event of a DBA.

In MODES 5 and 6 the possibility and consequences of a SLB or LOCA are low due to the RCS pressure and temperature limitations of these MODES. In addition, the CSS is not required to be available in these MODES.

The note states that the lower limit may be reduced to 60°F in MODES 2, 3, and 4 for both the upper and lower compartments.

ACTIONS

A.1

With containment average air temperature in the upper or lower compartment outside the subject LCO limits, it must be restored to within its limits within the Completion Time of 8 hours or else action must be taken to reduce power. This action must be taken to return the unit to within the bounds of the containment design bases analyses. The Completion Time of 8 hours is sufficient to correct minor problems or to prepare the unit for an orderly shutdown and entry into Condition B.

B.1 and B.2

The unit must be placed in à MODE in which the LCO does not apply if the containment average air temperature cannot be restored to within its limits within the allowed Completion Time of 8 hours. This is done by placing the unit in at least MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed to reach MODE 3 is a reasonable time, based on operating experience, to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 30 hours to reach MODE 5 from MODE 3 is a reasonable time considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5, maintaining containment average air temperature is no longer required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1 and SR 3.6.8.2

Verifying that the containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed in the containment design bases analyses. In order to determine the containment average air temperature, an [weighted] average is calculated using measurements taken at a minimum of [6] of the following specified locations within the containment:

Upper Compartment (SR 3.6.8.1)

a. El. 766 ft.
b. El. 809 ft.
c. El. 868 ft.

Lower Compartment (SR 3.6.8.2)

a. El. 745 ft.
b. El. 723 ft.
c. El. 708 ft. or
d. El. 726 ft.

These locations were selected to be representative of the overall containment atmosphere. The frequency of 24 hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".

2. Watts Bar FSAR, Section [6].

3. Watts Bar FSAR, Section [15].

4. Watts Bar FSAR, Section [6].

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, United States Nuclear Regulatory Commission, February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Ice Bed

BASES

BACKGROUND

The ice bed consists of over [1,999,800] lbs of ice stored in baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature would ensure having an OPERABLE containment and reduce the release of fission product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1).

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal plant operation. The upper plenum area is used for surveillance and maintenance of the ice bed.

The ice is held in the ice bed, within the ice condenser, in baskets arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser

(continued)

BASES (continued)

BACKGROUND (continued)

causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the Containment Spray System (CSS), are adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return Fan System (ARFS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

As the ice melts it passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to be a large source of borated water (via the containment sump) for long term Emergency Core Cooling System (ECCS) and CSS heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a DBA. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the CSS. The ice is adjusted to an alkaline pH which facilitates removal of radioactive iodine from the containment atmosphere. The alkaline pH also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the ECCS and CSS fluids in the recirculation mode of operation.

(continued)

BASES (continued)

BACKGROUND (continued)

It is important for the ice to be uniformly distributed around the 24 ice condenser bays, and for open flow paths to exist around the ice baskets. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays.

Two phenomena that can degrade the ice bed during the long service period are, 1) loss of ice by melting or sub-limitation, and 2) obstruction of flow passages through the ice bed due to buildup of frost or ice. Both of these degrading phenomena are reduced by maximizing a stablized ice bed temperature and minimizing air leakage into and out of the ice condenser.

The ice bed ensures the OPERABILITY of the containment by limiting the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment. Loss of containment integrity could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ice bed ensures the OPERABILITY of containment by limiting the pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to an OPERABLE containment are the Loss of Coolant Accident (LOCA) and the Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively.

Although the ice condenser is a passive system that requires no electrical power to perform its function, the CSS and the ARFS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, [in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, resulting in one train of CSS, RHR spray, EGTS, and ARFS to be inoperable].

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the [LOCA] analysis,

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and [is calculated to be less than the containment design temperature.] Therefore it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the limiting DBA SLB.

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice bed is a system that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

This LCO establishes the minimum requirements to assure that the ice bed is OPERABLE and will function to mitigate a containment DBA. The following requirements are included:

a. The ice bed temperature must be maintained well below the melting temperature to limit ice loss by melting,

b. The stored ice must have a boron concentration of at least [1800] ppm of sodium tetraborate in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation.

(continued)

BASES (continued)

LCOs (continued)

Sodium tetraborate is used since it has proven effective in maintaining the boron content for long storage periods. Sodium tetraborate also enhances the ability of the solution to remove and retain fission product iodine,

c. A high pH, $\geq$ [9.0] and $\leq$ [9.5] at 20 degrees C , is required to enhance the effectiveness of the ice and the melted ice in removing fission product iodine from containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the ECCS and CSS fluids in the recirculation mode of operation,

d. The minimum ice mass of [1,999,800] lbs includes a [11] % margin above the mass assumed in the safety analyses. This margin accounts for uncertainties in measurement and possible ice losses between surveillance periods.

e. The azimuthal distribution is important to assure that the ice is uniformly distributed around the ice condenser bays. This is especially important during the initial blowdown so that the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays,

f. Open flow channels through the ice bed and the ice condenser structures are required to assure that the steam and lower containment atmosphere will pass through and contact the ice, and

g. The total number of ice baskets in service assures that the ice mass is adequate and is distributed properly around the ice condenser bays.

APPLICABILITY

LCO 3.6.9 requires the ice bed OPERABLE to function to protect the integrity of the containment by limiting the pressure and temperature that could be experienced following a DBA. The ice condenser, together with the CSS and ARFS, provides the required heat removal capability to limit containment pressure and temperature during such an event.

(continued)

BASES (continued)

APPLICABILITY (continued)

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice bed. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, the ice bed is not required to be OPERABLE in these MODES.

ACTIONS

A.1

With the ice bed inoperable, the ice bed must be restored to OPERABLE status within the Completion Time of 48 hours before action must be taken to reduce power. The large ice mass and the ice condenser insulation assures that the ice bed will not degrade rapidly, and is a reasonable time for many repairs. As such, the Completion Time of 48 hours is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

B.1 and B.2

If the inoperable ice bed cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5, the ice bed is no longer required OPERABLE to protect the integrity of containment due to the pressure and temperature limitations of this MODE.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Verifying that the maximum temperature of the ice bed is ≤ [27]°F ensures that the ice is kept well below the melting point. Because of the large ice mass the temperature is not likely to change rapidly. Therefore the frequency of 12 hours was established based on engineering judgment, and has been shown to be acceptable through industry operating experience. This surveillance may be satisfied by use of the Ice Bed Temperature Monitoring System.

SR 3.6.9.2

Verifying the chemical composition of the stored ice ensures that the stored ice has a boron concentration of at least [1800] ppm of sodium tetraborate and a high pH, ≥ [9.0] and ≤ [9.5] at 20 degrees C, in order to meet the requirement for borated water when the melted ice is used in the ECCS recirculation mode of operation. Sodium tetraborate is used since it has proven effective in maintaining the boron content for long storage periods. Sodium tetraborate also enhances the ability of the solution to remove and retain fission product iodine. The high pH is required to enhance the effectiveness of the ice and the melted ice in removing iodine from containment atmosphere. This pH range also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the ECCS and CSS fluids in the recirculation mode of operation. Long term ice storage tests have determined that the chemical composition of the stored ice is extremely stable. The specification of at least 9 representative samples and the frequency of 9 months were based on engineering judgment and have been shown to be acceptable through industry operating experience.

SR 3.6.9.3

The weighing program is designed to obtain a representative sampling of the ice baskets. The representative sample shall include 6 baskets from each of the 24 ice condenser bays, and consist of one basket from radial rows 1, 2, 4, 6, 8, and 9. If no basket from a designated row can be obtained for weighing, a basket from the same row of an adjacent bay shall be weighed. The rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1, 2, and 8, 9 respectively) where heat

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.3 (continued)

transfer into the ice condenser is most likely to influence melting or sublimation. Verifying the total weight of ice assures that there is adequate ice to absorb the required amount of energy to mitigate the DBAs. If a basket is found to contain < [1029] lbs of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. the average weight of ice in these 21 baskets shall be ≥ [1029] lbs at a [95]% confidence level. Weighing 20 additional baskets from the same bay in the event a single basket contains < [1029] lbs assures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt out during a DBA transient, creating a path for steam to pass through the ice bed without being condensed. The frequency of 9 months was based on ice storage tests and the margin built into the required ice mass over and above the mass assumed in the safety analyses, and has been shown to be acceptable through industry operating experience.

SR 3.6.9.4

This surveillance assures that the azimuthal distribution of ice is reasonably uniform by verifying the average ice weight in three azimuthal groups of ice condenser bays are within limits. The surveillance interval of 18 months is based on engineering judgment, and ice bed operational experience.

SR 3.6.9.5

This surveillance ensures that the flow channels through the ice condenser have not accumulated an excessive amount of ice or frost blockage. The allowable [0.38] inch thick buildup of frost or ice is based on analysis of containment response to a DBA with partial blockage of the ice condenser flow passages. If a flow channel in a given bay is found to have an accumulation of frost or ice > [0.38] inches thick, a representative sample of 20 additional flow channels from the same bay shall be visually inspected. If these additional flow channels are all found to be acceptable, the discrepant flow channel may be considered as a single unique and acceptable deficiency. However, more than one discrepant flow channel in a bay is not acceptable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.5 (continued)

The visual inspection shall be made for $\geq$ 2 flow channels per ice condenser bay, and shall include the following specific locations along the flow channel:

a. Past the lower inlet plenum support structures and turning vanes,

b. Between ice baskets,

c. Past lattice frames,

d. Through the intermediate floor grating, and

e. Through the top deck floor grating.

The use of 20 additional flow paths to ascertain if a single discrepant flow path is acceptable is based on engineering judgment. The frequency of 9 months is based on engineering judgment, and has been shown to be acceptable through operating experience.

SR 3.6.9.6

This surveillance ensures that a representative sampling of ice baskets ,which are relatively thin-walled perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket shall be raised at least 12 feet for this inspection. The surveillance interval of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment, and has been shown to be acceptable through operating experience.

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.

2. Watts Bar FSAR, Section [6].

3. NRC Interim Policy Statement, 52FR3788, "Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Ice Condenser Doors

BASES

BACKGROUND

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are (1) to seal the ice condenser from air leakage during the lifetime of the unit, and (2) to open in the event of a Design Basis Accident (DBA) to direct the hot steam/air mixture from the DBA into the ice bed where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature would ensure containment OPERABILITY and reduce the release of fission product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1).

The ice condenser is an annular compartment enclosing approximately 300 degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. The upper plenum area is used for surveillance and maintenance of the ice bed.

The ice is held in the ice bed, within the ice condenser, in baskets arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors

(continued)

BASES (continued)

BACKGROUND (continued)

to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildup in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the Containment Spray System (CSS), are adequate to absorb the initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return Fan System (ARFS) returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The ice condenser doors assure that the ice stored in the ice bed is preserved during normal operation (doors closed), and that the ice condenser functions as designed if called upon to act as a massive heat sink following a DBA. As such the ice condenser doors protect the integrity of the containment by limiting the pressure and temperature that could be expected following a DBA. Containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The ice condenser doors ensure the integrity of containment by limiting the pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the Loss of Coolant Accident (LOCA) and the Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Although the ice condenser is a passive system that requires no electrical power to perform its function, the CSS and the ARFS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, [in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, resulting in one train of CSS, RHR spray, EGTS, and ARFS to be inoperable].

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the [LOCA] analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and [is calculated to be less than or equal to the containment design temperature.] Therefore it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the limiting DBA SLB.

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The ice condenser doors are components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, they satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCOs

This LCO establishes the minimum equipment requirements to ensure that the ice condenser doors perform their safety functions. The ice condenser inlet doors, intermediate deck doors, and top deck doors must be closed to minimize air leakage into and out of the ice condenser, with its attendant leakage of heat into the ice condenser and loss of ice through melting and sublimation. The doors must be OPERABLE to ensure the proper opening of the ice condenser in the event of a DBA. OPERABILITY includes being free of any obstructions that would limit their opening, and, for the inlet doors, being adjusted such that the opening and closing torques are within limits. The ice condenser doors protect the integrity of the containment by functioning with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

LCO 3.6.10 requires the ice condenser doors OPERABLE to function to protect the integrity of the containment by limiting the pressure and temperature that could be experienced following a DBA. The ice condenser, together with the CSS, RHR spray as necessary, and ARFS, provides the required heat removal capability to limit containment pressure and temperature during such an event.

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the operation of the ice condenser doors. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, the ice condenser doors are not required OPERABLE in these MODES.

ACTIONS

A.1

If one or more inlet doors are physically restrained from opening, the unit is not prepared to respond to the design basis events for which this system is required. A Completion Time of 12 hours is allowed to restore the door to OPERABLE status before action must be taken to reduce power. The specified Completion Time is sufficient to allow the correction of minor problems or prepare the unit for an orderly shutdown. As such, the 12 hour Completion Time is based on engineering judgment.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If one or more are determined to be partially open or otherwise inoperable (except for Condition A), it is acceptable to continue unit operation for up to 14 days provided the ice bed temperature instrumentation is monitored to assure that the open or inoperable door is not allowing enough air leakage to cause the maximum ice bed temperature to approach the melting point. The frequency of 4 hours is based on the fact that temperature changes cannot occur rapidly, and if the ice bed temperature is > [27] °F at any time, the situation reverts to Condition C, and a Completion Time of 48 hours is allowed to restore the inoperable door to OPERABLE status or enter into Required Actions D.1 and D.2. Provided the temperature is maintained below the limit, the Completion Time of 14 days to restore the door is based on engineering judgment.

C.1

If Required Actions B.1 or B.2 are not met, the large mass of ice assures that the ice bed will not degrade rapidly. The requirement to restore the doors to OPERABLE status within the Completion Time of 48 hours is based on engineering judgment and has been shown to be acceptable through industry operating experience.

D.1 and D.2

If the Required Actions are not met within the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In MODE 5 the ice condenser is no longer required OPERABLE to protect the integrity of containment due to the RCS pressure and temperature limitations of this MODE.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Verification of the inlet doors in their closed positions by the Inlet Door Position Monitoring System causes operator awareness of an inadvertent opening of one or more doors. The surveillance frequency of 12 hours is based on assuring that operators on each shift are aware of the status of the doors.

SR 3.6.10.2

Verifying by visual inspection that each intermediate door is closed and not impaired by ice, frost, or debris, provides assurance that the intermediate deck doors (which form the floor of the upper plenum where frequent maintenance on the ice bed is performed) have not been left open or obstructed. The frequency of 7 days is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.10.3

Verifying by visual inspection that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a LOCA or SLB. The frequency of 18 months is acceptable due to ice bed operating experience that ice and frost buildup are not a significant problem. Because of high radiation in the vicinity of the inlet doors during power operation, this surveillance is normally performed during a shutdown.

SR 3.6.10.4

Demonstrate the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of [675] inch-lbs is based on the design opening pressure on the doors [ ] lb/sq. ft. The frequency of 18 months is acceptable due to extended ice bed operating experience that sticking of doors is not a significant problem. Because of high radiation in the vicinity of the inlet doors during power operation, this surveillance is normally performed during a shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.5

The torque test surveillance assures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The purpose of the return torque specification is to assure that in the event of a small break LOCA, or SLB, all of the 24 door pairs open uniformly. This assures that during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. The frequency of 18 months has been determined acceptable based on engineering judgement of ice bed operating experience. Because of high radiation in the vicinity of the inlet doors during power operation, this surveillance is normally performed during a shutdown.

The sampling of inlet doors for the torque test shall be selected such that all inlet doors are tested at least once every 4 test intervals.

For each inlet door in the sampling:

a. Demonstrate torque, T(open), required to cause opening motion at the [40]° open position is ≤[195] inch-lbs.

b. Demonstrate torque, T(close), required to hold the door stationary (i.e., keep it from closing) at the [40]° open position is ≤[78] inch-lbs, and

c. Demonstrate frictional torque, T(frict) = 0.5 {T(open)-T(close)} is ≤ [40] inch-lbs.

SR 3.6.10.6

Verifying the OPERABILITY of the intermediate deck doors, including verification of free movement of the vent assemblies and the door lifting forces, provides assurance that the intermediate deck doors are free to open in the event of a DBA. The frequency of 18 months is based on extended ice bed operating experience that has demonstrated that these areas are not a signficant problem.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.10.6 (continued)

Demonstrate free movement of the door with lifting forces as follows:

Door	Lifting Force	Orientation
[01,05]	≤ [38] lbs	Adjacent to crane wall
[02,06]	≤ [35] lbs	Paired with door adj. to crane wall
[03,07]	≤ [33] lbs	Adjacent to containment wall
[04,08]	≤ [32] lbs	Paired with door adj. to containment wall

SR 3.6.10.7

Verifying by visual inspection that the top deck doors are in place and are not obstructed provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation and would not be obstructed if called upon to open in response to a DBA. The frequency of 92 days is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.

2. Watts Bar FSAR, Section [6].

3. NRC Interim Policy Statement, 52FR3788, "Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 Divider Barrier Integrity

BASES

BACKGROUND

The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a Design Basis Accident (DBA). This ensures that most of the gases pass through the ice bed, which condenses the steam and limits pressure and temperature during the accident transient. Limiting the pressure and temperature would ensure containment OPERABILITY and reduce the release of fission product radioactivity from containment to the environment in the event of a DBA to less than the guidelines of 10 CFR 100 (Ref. 1).

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the condenser to open, which allows the air to flow out of the ice condenser into the upper compartment. The ice condenses the steam as it enters, thus limiting the pressure and temperature buildups in containment. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the Containment Spray System (CSS), are adequate to absorb initial blowdown of steam and water from a DBA and the additional heat loads that would enter containment during several hours following the initial blowdown. The additional heat loads would come from the residual heat in the reactor core, the hot piping and components, and the secondary system, including the steam generators. During the post-blowdown period, the Air Return Fan System (ARFS) returns upper compartment air through

(continued)

BASES (continued)

BACKGROUND (continued)

the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam from the lower compartment through the ice condenser, where the heat is removed by the remaining ice.

The divider barrier integrity assures that the high energy fluids released during a DBA would be directed through the ice condenser, and that the ice condenser would function as designed if called upon to act as a massive heat sink following a DBA. As such the divider barrier ensures the integrity of the containment by limiting the pressure and temperature that could be expected following a DBA. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The divider barrier integrity ensures the integrity of containment by limiting the pressure and temperature that could be experienced following a DBA. The limiting DBAs considered relative to containment OPERABILITY are the Loss of Coolant Accident (LOCA) and the Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the CSS and the ARFS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, [in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, resulting in one train of CSS, RHR spray, EGTS, and ARFS to be inoperable].

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the [LOCA] analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and [is calculated to be less than or equal to the containment design temperature.]

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Therefore it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the limiting DBA SLB.

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

The divider barrier is a structure that is part of the primary success path which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

This LCO establishes the minimum equipment requirements to assure that the divider barrier performs its safety function of assuring that the bypass leakage during DBA does not exceed the bypass leakage assumed in the accident analysis. Included are the requirements that the personnel access doors and equipment hatches in the divider barrier are OPERABLE and closed, and that the divider barrier seal is properly installed and has not degraded with time. An exception to the requirement that the doors be closed is made to allow personnel transit entry through the divider barrier. The basis of this exception is the assumption that, for personnel transit, the time during which a door is open will be short, i.e., shorter than the Completion Time of 1 hour for Condition A. The divider barrier protects the integrity of the containment by functioning with the ice condenser to limit the pressure and temperature that could be expected following a DBA.

(continued)

BASES (continued)

APPLICABILITY

LCO 3.6.11 requires divider barrier integrity to protect the integrity of the containment by assuring the ice condenser functions to limit the pressure and temperature that could be experienced following a DBA. The ice condenser, together with the CSS and ARFS, provides the required heat removal capability to limit containment pressure and temperature during such an accident.

In MODES 1, 2, 3, and 4, a DBA could cause an increase in containment pressure and temperature requiring the integrity of the divider barrier. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, divider barrier integrity is not required in these MODES to protect containment OPERABILITY.

ACTIONS

A.1

If one or more personnel access doors or equipment hatches are inoperable or open, except for personnel transit entry, the unit is not fully prepared to respond to the design basis events for which the divider barrier integrity is required. One hour is allowed to restore the door to OPERABLE status and the closed position before action must be taken to reduce power. The specified Completion Time is sufficient to allow the correction of minor problems or prepare the unit for an orderly shutdown and entry into Required Actions C.1 and C.2. As such, the Completion Time of 12 hours is based on engineering judgment.

B.1

If the divider barrier seal is inoperable, the plant is not fully prepared to respond to the design basis events for which the divider barrier integrity is required. Twelve hours is allowed to restore the seal to OPERABLE status before action must be taken to reduce power. The specified Completion Time is sufficient to allow the correction of minor problems or prepare the unit for an orderly shutdown and entry into Required Actions C.1 and C.2. As such, the Completion Time of 12 hours is based on engineering judgment.

(continued)

BASES (continued)

ACTIONS (continued)

C.1 and C.2

If the Required Actions are not met within the required Completion Time, the plant must be placed in a MODE where the requirement does not apply. This is performed by placing the plant in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In Mode 5 the ice condenser is no longer required OPERABLE to protect the integrity of containment due to the RCS pressure and temperature limitations of this MODE.

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

Verification by visual inspection that all personnel access doors and equipment hatches between the upper and lower containment compartments are closed provides assurance that the divider barrier integrity is maintained during each startup prior to the reactor being taken from MODE 5 to MODE 4. This SR is necessary because many of the doors and hatches may have been opened for maintenance during the shutdown.

SR 3.6.11.2

Verification that the personnel access door and equipment hatch seals, sealing surfaces, and alignments are acceptable provides assurance that the divider barrier integrity is maintained. This inspection cannot be made when the door or hatch is closed. Therefore, SR 3.6.11.2 is required for each door or hatch that has been opened, prior to the final closure. Some doors and hatches may not be opened for long periods of time. Those that use resilient materials in the seals must be opened and inspected at least once per 10 years to provide assurance that the seal material has not aged to the point of degraded performance. The frequency of 10 years is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.3

Verification after each opening of a personnel access door or equipment hatch that it has been closed causes operator awareness of the importance for closing it, and thereby provides additional assurance that the divider barrier integrity is maintained while in applicable MODES.

SR 3.6.11.4

Conducting periodic physical property tests on the divider barrier seal test coupons provides assurance that the seal material (typically [2 ply dacron coated EDPM]) has not degraded in the containment environment, including the effects of irradiation with the reactor at power. The required tests include a tensile strength test. If a failure occurs in the initial testing, a larger sample is taken to test at reduced pressure. If a failure occurs again, the samples must be sent to the manufacturer, for LOCA environmental simulation testing involving radiation humidity and temperature to see if the seal is still OPERABLE. The frequency of 18 months is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.11.5

Visual inspection of the seal around the perimeter provides assurance that the seal is properly secured in place. The frequency of 18 months is based on engineering judgment, and has been shown to be acceptable through operating experience.

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.
2. Watts Bar FSAR, Section [6].
3. NRC Interim Policy Statement, 52FR3788, "Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.12 Hydrogen Analyzer System

BASES

BACKGROUND

The Hydrogen Analyzer System is a subsystem of the Combustible Gas Control System (CGCS) , and is part of the Engineered Safety Features of the plant. The CGCS is designed to control the hydrogen concentration that may accumulate in containment following a Design Basis Accident (DBA). The CGCS consists of:

a. The Hydrogen Analyzer System (HAS), which measure the hydrogen concentration in containment so that operator action may be taken to control the bulk hydrogen concentration below the flammable concentration,

b. The Hydrogen Recombiner System (HRCS), which removes hydrogen from the containment atmosphere by raising the temperature above the ignition temperature for combustion of the hydrogen/air mixture in the recombiner, and

c. The hydrogen collection system which functions in conjuction with the Air Return Fan System to take suction from dead-ended compartments to prevent hydrogen buildup.

Each of the two HASs consists of a hydrogen analyzer and associated sample lines from containment. The sample lines have isolation valves inside containment. The analyzers are located in accessible areas outside containment. Each hydrogen analyzer is powered from a separate Engineered Safety Features bus. During normal operation the hydrogen analyzers are isolated from the containment atmosphere.

Emergency Operating Procedures direct that the hydrogen concentration in containment be monitored. To begin hydrogen monitoring, the operator would manually open the isolation valves and initiate hydrogen sampling. Individual valve control switches are provided in the main control room with a provision for remote manual bypass. Containment atmosphere samples, maintained in vapor phase, are brought to the monitor, which measures the concentration of hydrogen. The sample is then returned to the containment atmosphere. The hydrogen analyzer is designed with the

(continued)

BASES (continued)

BACKGROUND (continued)

capability to obtain an accurate sample 30 minutes after initiation of safety injection. The analyzers, when actuated, will continuously monitor hydrogen concentration levels ≥ [0] and ≤ [10] volume percent (v/o). Each analyzer draws a sample from areas that have been selected to provide a representative sample of the containment atmosphere following a hydrogen release in containment.

The hydrogen analyzers protect the integrity of the containment by providing the operator with the capability to monitor the hydrogen concentration in containment following a DBA, and to take the action necessary to prevent the occurrence of a bulk hydrogen burn inside containment. Maintenance of containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CGCS protects the integrity of containment by providing the capability of controlling the bulk hydrogen concentration in containment to less than the minimum flammable concentration of 4.0 v/o (Ref. 2), following a DBA. This control would prevent a containment-wide hydrogen burn, thus ensuring containment OPERABILITY and minimizing challenges to the OPERABILITY of safety-related equipment located in containment. The limiting DBA relative to hydrogen generation is a Loss of Coolant Accident (LOCA).

The CGCS design begins with consideration of the limiting DBA 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. This reaction does not occur at a significant rate unless local clad temperatures in excess of 1800°F exist,

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, and

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions. Note that containment spray may also occur following a Steam Line Break (SLB). In fact, corrosion is the only significant source of hydrogen following a SLB. For this reason, a SLB is much less limiting than a LOCA from a hydrogen generation standpoint.

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. As such, the HCS is designed to control an amount of hydrogen generation in containment considerably in excess of the amount that would be calculated from the limiting DBA LOCA. For example, the two largest sources of hydrogen (items a. and b. above) are increased by a factor of 5 or more from the amounts that would be calculated from the limiting DBA LOCA.

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 4.0 v/o.

The hydrogen analyzers provide the operator with the capability to measure the hydrogen concentration in containment. This capability is necessary to confirm that the hydrogen recombiners are operating properly to maintain the bulk hydrogen concentration below the minimum flammable concentration following a LOCA or containment high pressure condition. In the unlikely event that the recombiners are not successful, monitoring would allow the operator to anticipate the hydrogen concentration buildup and initiate backup action (such as containment purge) to minimize the likelihood of a bulk hydrogen burn inside containment. As such, the hydrogen analyzers ensure that containment OPERABILITY is maintained. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The hydrogen analyzers are part of the primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, they satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

This LCO establishes the minimum equipment required OPERABLE for the hydrogen monitoring function following accidents which can involve generation of hydrogen gas in containment. Two [independent and redundant] hydrogen analyzers are required to ensure that one is available in the event the other fails to operate.

APPLICABILITY

The requirement for hydrogen analyzers in MODES 1 and 2 is based on a highly conservative analysis of potential hydrogen production (in excess of the amount that would be predicted from a DBA LOCA analysis). This conservative analysis indicates that unless controlled by the HCS, the bulk hydrogen concentration would build up over several days time after the initiating event and would reach the 4.0 v/o minimum flammable concentration in approximately [ ] days (Ref. 4). Thus, in the first few days after the accident, hydrogen monitoring would be necessary to confirm the effectiveness of the HCS and to make decisions whether any backup actions are needed. For this reason the hydrogen monitors LCO is applicable in MODES 1 and 2.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less (by 20 to 40%) than that calculated for the DBA LOCA (the starting point for the conservative hydrogen analysis). Thus, if the hydrogen analysis were to be performed starting with a LOCA in MODE 3 or 4, the time to reach a bulk concentration of 4 v/o would be extended beyond the [ ] days conservatively calculated for the MODES 1 and 2. The extended time would allow containment atmosphere sampling by other means to determine the hydrogen buildup, if the hydrogen monitors were not available. Therefore the hydrogen monitors are not required to be OPERABLE in MODES 3 and 4. In MODE 5 they are not required because containment OPERABILITY is not required.

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BASES (continued)

APPLICABILITY (continued)

The NOTE stating that LCO 3.0.4 is not applicable allows MODE changes while the hydrogen analyzers are in an inoperable condition since the hydrogen analyzers in no way affect operation in MODES 1 and 2.

ACTIONS

A.1

With one hydrogen analyzer inoperable, the inoperable hydrogen analyzer must be restored to OPERABLE status within the Completion Time of 30 days before action must be taken to reduce power. The Completion Time of 30 days is based on the following considerations:

a. The low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the minimum flammable concentration,

b. The OPERABILITY of the other hydrogen analyzer and

c. The availability of the hydrogen recombiners which can be activated independently of the hydrogen monitors following a DBA if there is reason to suspect hydrogen is being generated in containment.

B.1

With two hydrogen analyzers inoperable, at least one inoperable analyzer must be restored to OPERABLE status

within the Completion Time of 7 days before action must be taken to reduce power. The Completion Time of 7 days is based on the following considerations:

a. The low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the minimum flammable concentration,

b. The length of time ([9] days) after the event before hydrogen would build up to a minimum flammable concentration, during which containment atmosphere could be sampled for hydrogen concentration by other means,

(continued)

BASES (continued)

ACTIONS (continued)

B.1 (continued)

c. The availability of the hydrogen recombiners which can be activated independently of the hydrogen monitors following a DBA if there is reason to suspect hydrogen is being generated in containment, and

d. The NRC specifically recommended 7 days in their TMI action plan document (Ref. 5).

C.1

If the inoperable hydrogen analyzers cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is performed by placing the unit in MODE 3 in 6 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. In MODE 3 the hydrogen analyzers are no longer required OPERABLE to protect the integrity of containment due to the reduced potential hydrogen production rates for a LOCA in this MODE.

SURVEILLANCE REQUIREMENTS

SR 3.6.12.1

Performing a CHANNEL CHECK provides assurance that the channels have not drifted outside their limits. A CHANNEL CHECK is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two channels of indication should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the electronics have drifted outside of their limits. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their trip setpoints. The frequency of 12 hours is based on engineering judgment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.2

The performance of an ANALOG CHANNEL OPERATIONAL TEST provides assurance that the analog process control equipment is within limits. This test is a periodic check of the process control equipment. When the channel is placed in the test condition, the input from the transmitter is removed and the alarm trip output is isolated. This allows a test signal to be introduced into the instrument loop. The input can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The alarm trip setpoint of the channel can be determined by varying the input and observing the trip status. The frequency of 31 days is based on engineering judgment, and has been shown to be acceptable through operating experience.

SR 3.6.12.3

Performance of a CHANNEL CALIBRATION on the hydrogen analyzers ensures the accuracy of the monitors by calibrating with a gas sample of known hydrogen concentration. The sample gases used for performing the CHANNEL CALIBRATION are nominal [1.0] v/o hydrogen, balance nitrogen, and nominal [4.0] v/o hydrogen, balance nitrogen. Since the hydrogen concentration limit has been established as 4.0 v/o hydrogen in air or steam-air atmospheres (Ref. 1), calibration with these sample gases ensures the accuracy of the hydrogen monitors over the range of interest. The frequency of 92 days on a STAGGERED TEST BASIS for this surveillance is based on engineering judgment and has been shown to be acceptable through operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.

2. Regulatory Guide 1.7, Rev. 2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident", United States Nuclear Regulatory Commission, November 1978.

3. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications", May 9, 1988.

4. Watts Bar FSAR, Section [6.2.5]

5. "NRC Staff Guidance on Technical Specifications Requirements for NUREG 0737 Items Scheduled After December 31, 1981".

B 3.6 CONTAINMENT SYSTEMS

B 3.6.13 Hydrogen Recombiner System (HRS)

BASES

BACKGROUND

The hydrogen recombiners are a subsystem of the Combustible Gas Control System (CGCS) [, and are part of the Engineered Safety Features of the plant]. The CGCS is designed to control the hydrogen concentration that may accumulate in containment following a Design Basis Accident (DBA). The CGCS consists of:

a. The hydrogen monitors, which measure the hydrogen concentration in containment, so that operator action may be taken to control the bulk hydrogen concentration below the flammable concentration,

b. The hydrogen recombiners, which remove hydrogen from the containment atmosphere by raising the temperature above the ignition temperature for combustion of the hydrogen/air mixture in the recombiner, and

c. The hydrogen collection system which functions in conjuction with the Air Return Fan System to take suction from dead-ended compartments to prevent hydrogen buildup.

Each hydrogen recombiner consists of a control panel and a power supply cabinet located in the control building, and a recombiner unit located inside containment. The recombiner unit is a passive system having no moving parts, and requiring only electrical power to initiate its function. The power supply and controls are manually actuated.

The containment atmosphere is heated in the recombiner in a vertical duct, causing it to rise by natural convection. As it rises, replacement air is drawn into the intake louvers downward through a preheater section. The preheated air then flows through an orifice plate, sized to

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BASES (continued)

BACKGROUND (continued)

maintain the desired flow rate, to the heater section where it is heated to a temperature above 1150°F, the minimum temperature for combustion of the hydrogen/air mixture in the recombiner. The hydrogen burns, converting it into water vapor and, after passing through the combustion section, the hot gases enter a mixing section where they are mixed and cooled with containment atmosphere before being discharged back into containment.

A single recombiner is capable of meeting the safety requirements, and two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate [Engineered Safety Feature] bus, and is provided with a separate power panel and control panel.

Emergency Operating Procedures direct that the hydrogen concentration in containment be monitored following a DBA, and that the hydrogen recombiners be manually activated to prevent the containment atmosphere from reaching a bulk hydrogen concentration of 4.0 volume percent (v/o).

The hydrogen recombiners protect the integrity of the containment by providing the operator with the capability of preventing the occurrence of a bulk hydrogen burn inside the containment. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CGCS protects the integrity of containment by providing the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o (Ref. 2), following a DBA. This control would prevent a containment-wide hydrogen burn, thus ensuring containment OPERABILITY and minimizing challenges to the OPERABILITY of safety-related equipment located in containment. The limiting DBA relative to hydrogen generation is a Loss of Coolant Accident (LOCA).

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The CGCS design begins with consideration of the limiting DBA 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. This reaction does not occur at a significant rate unless local clad temperatures in excess of 1800°F exist,

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, and

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions. Note that containment spray may also occur following a Steam Line Break (SLB). In fact, corrosion is the only significant source of hydrogen following a SLB. For this reason, a SLB is much less limiting than a LOCA from a hydrogen generation standpoint.

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. As such, the CGCS is designed to control an amount of hydrogen generation in containment considerably in excess of the amount that would be calculated from the limiting DBA LOCA. For example, the two largest sources of hydrogen (items a. and b. above) are increased by a factor of 5 or more from the amounts that would be calculated from the limiting DBA LOCA.

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 4.0 v/o.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	The hydrogen recombiners provide operators with the ability to prevent occurrence of a bulk hydrogen burn inside the Containment. As such they help ensure containment OPERABILITY is maintained. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1). Hydrogen recombiners are components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, they satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 4).
LCOs	This LCO establishes the minimum equipment required OPERABLE for the hydrogen recombination safety function following accidents which can involve generation of hydrogen gas in containment. Two hydrogen recombiners are required to ensure that at least one is available assuming the other one fails to operate.
APPLICABILITY	The requirement for hydrogen recombiners in MODES 1 and 2 is based on a highly conservative analysis of potential hydrogen production (in excess of the amount that would be predicted from a DBA LOCA analysis). This conservative analysis indicates that unless controlled by the CGCS, the bulk hydrogen concentration would build up over several days time after the initiating event and would reach the 4.0 v/o minimum flammable concentration in approximately [9] days (Ref. 4). In such an event the recombiners must be OPERABLE, if needed to control the bulk hydrogen concentration in containment below the flammable concentration. For this reason the hydrogen recombiner LCO is applicable in these MODES. The LOCA analyses predict fuel failures only if the accident occurs while in MODES 1 or 2, and in MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be significantly less (by 20 to 40%) than that calculated for the DBA LOCA. This means that the time to reach a bulk concentration of 4 v/o (already highly conservative, as discussed above) would be

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BASES (continued)

APPLICABILITY (continued)

extended beyond the calculational time frame of a DBA. This, in itself, would not necessarily preclude the need for some means of hydrogen control. However, since no fuel failures are predicted for a LOCA in MODES 3 or 4, there would be very little fission product release to the containment atmosphere. Furthermore, the extended time would allow additional time for radioactive decay of any non-fuel airborne fission products in the containment atmosphere. These factors lead to the conclusion that the Containment Hydrogen Purge System could be used, in lieu of the recombiners, without significantly effecting the environmental consequences of the event, and, therefore the hydrogen recombiners are not required to be OPERABLE in MODES 3 and 4. In MODE 5 they are not required because containment OPERABILITY is not required.

The NOTE stating that LCO 3.0.4 is not applicable allows MODE changes while the hydrogen recombiners are in an inoperable condition since the hydrogen recombiners in no way affect normal operation in MODES 1 and 2.

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within the Completion Time of 30 days before action must be taken to reduce power. The Completion Time of 30 days is based on the following considerations:

a. The low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammable concentration, and

b. The OPERABILITY of the other hydrogen recombiner.

(continued)

BASES (continued)

ACTIONS (continued)

B.1

With two hydrogen recombiners inoperable, at least one inoperable recombiner must be restored to OPERABLE status within the Completion Time of 7 days before action must be taken to reduce power. The Completion Time of 7 days is based on the following considerations:

a. The low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the minimum flammable concentration,

b. The length of time ([9] days) after the event before hydrogen could exceed the flammability concentration (longer than the Completion Time of 7 days).

C.1

If the inoperable hydrogen recombiners cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. In MODE 3 the hydrogen recombiners are no longer required OPERABLE to protect the integrity of containment due to the reduced potential hydrogen production rates and virtually zero potential for fission product release to containment for a LOCA in this MODE.

SURVEILLANCE REQUIREMENTS

SR 3.6.13.1

SR 3.6.13.1 is the performance of a functional test. The functional test verifies that the recombiner is capable of producing the internal temperature level required to cause combustion of hydrogen in air. The temperature required is 1150°F. However, periodic testing at this temperature would degrade the recombiner by shortening the lifetime of the resistance heaters. To avoid this undesirable result, the functional test is performed at a lower temperature. The recombiner is activated and the [internal] temperature increase to at least [700]°F within 90 minutes verifies that the heaters are operating.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.13.1 (continued)

The second step is to increase the power setting to maximum and verify the total power. This confirms that all three power phases are operating, and assures that the unit is capable of achieving the desired operating temperature. If a single phase is not operating, the heaters will not be capable of achieving the desired temperature. However, the maximum power would be reduced by an easily-detectable 33 %. The frequency of 6 months is based on engineering judgment and has been shown to be acceptable through industry operating experience.

SR 3.6.13.2

SR 3.6.13.2 is the performance of a CHANNEL CALIBRATION of the hydrogen recombiner instrumentation and control circuits. A CHANNEL CALIBRATION is performed approximately every 18 months, and consists of a verification of the temperature and power instrumentation [and the automatic power controller]. The frequency of 18 months is based on engineering judgment and has been shown to be acceptable through industry operating experience.

SR 3.6.13.3

SR 3.5.18.3 is a visual examination of the recombiner to verify that there is no evidence of abnormal conditions within the recombiner enclosures. Since the recombiners operate on natural circulation flow of containment atmosphere, the visual inspection should confirm that the inlet and outlet louvers and the internal flow passages are clear of any accumulation of dust or debris. Examples of other abnormal conditions would be loose wiring or loose structural connections. The frequency of 18 months is based on engineering judgment and has been shown to be acceptable through industry operating experience.

SR 3.6.13.4

SR 3.6.13.4 is a verification of the integrity of the hydrogen recombiner electrical heaters by performing a resistance to ground test for each heater phase. The frequency of 18 months is based on engineering judgment and has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.

2. Regulatory Guide 1.7, Rev. 2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident", United States Nuclear Regulatory Commission, November 1978.

3. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications", May 9, 1988.

4. Watts Bar FSAR, Section [6.2.5]

B 3.6 CONTAINMENT SYSTEMS

B 3.6.14 Hydrogen Mitigation System

BASES

BACKGROUND

The Hydrogen Mitigation System (HMS) is designed to increase the containment capability to accommodate hydrogen that could be released during a degraded core accident. As such, the hydrogen igniters are designed to mitigate potential hydrogen release far in excess of the amount postulated to result from the limiting Design Basis Accident (DBA) Loss of Coolant Accident (LOCA) in accordance with 10 CFR 50.44 (Ref. 1). Such DBA analysis considers hydrogen equivalent to a zirconium - steam reaction involving 1.5 to 5.0% of the core cladding.

As a result of Nuclear Regulatory Commission rule making following the Three Mile Island accident 10 CFR 50.44 was amended to require ice condenser plants to install suitable hydrogen control systems which would accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water, without loss of containment OPERABILITY. The HMS provides this required capability. This requirement was imposed on ice condenser plants because of their small containment volume and low design pressure (compared with pressurized water reactor dry containments). Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, if ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the containment OPERABILITY and OPERABILITY of safety systems in containment.

The HMS is based on the concept of controlled ignition using thermal igniters, designed to be redundant, capable of functioning in a post-accident environment, seismically supported, and capable of actuation from the control room. A total of [68] igniters are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The igniters are arranged in two independent trains such that each containment region has at least 2 igniters, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize.

(continued)

BASES

BACKGROUND (continued)	The hydrogen igniters protect the integrity of containment, and minimize challenges to safety equipment located within containment, by limiting the temperatures and pressures that could be experienced from a hydrogen burn following a degraded core accident. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The HMS will function to cause hydrogen in containment to burn in a controlled manner as it accumulates following a severe accident. Burning would occur at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup. The hydrogen igniters are not included for mitigation of a DBA because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA LOCA. However, the hydrogen igniters have been shown by Probablistic Risk Analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for ice condenser plants. As such, the hydrogen igniters are considered to be risk significant in accordance with the NRC Interim Policy Statement (Ref. 3).
LCOs	LCO 3.6.14 establishes the minimum HMS equipment required to accomplish the hydrogen mitigation safety function following a degraded core accident. Two trains of the system are required OPERABLE to ensure that at least one is available assuming failure of the other train to actuate.

(continued)

BASES (continued)

APPLICABILITY

LCO 3.6.14 is applicable in MODES 1 and 2 based on potential hydrogen production following a degraded core accident. The possibility and consequences of such an event in MODES 3, 4, and 5 are low due to the power limitations of these MODES. As such, the hydrogen igniters are not required to be OPERABLE in these MODES to ensure containment OPERABILITY.

ACTIONS

A.1 and A.2

With one HMS train inoperable, the inoperable train must be restored to OPERABLE status within the specified Completion Time before action must be taken to reduce power. The Completion Time of 7 days is based on the low probability of a severe accident, and the OPERABLE status of the other train.

If the inoperable HMS train cannot be restored within the Completion time of 7 days, it is acceptable to continue operation provided SR 3.6.14.1 is performed on the OPERABLE train with the specified frequency of 7 days. This SR verifies at least one hydrogen igniter OPERABLE in each containment zone. The igniters are simple glow plug devices, not likely to fail between surveillance periods with a frequency of 7 days. As such, the frequency is based on engineering judgment, and has been shown to be acceptable through operating experience.

B.1

With both HMS trains inoperable, the unit is not prepared to respond to the degraded core event for which this system is required. The Completion Time of 7 days is allowed to restore one train of hydrogen igniters to OPERABLE status before action must be taken to reduce power: The specified Completion Time is sufficient to allow the correction of system problems or prepare the unit for an orderly shutdown and entry into Required Action C.1.

(continued)

BASES (continued)

ACTIONS (continued)

C.1

If the inoperable HMS trains cannot be restored in the required Completion Time, or the Required Actions of Condition A are not met, the unit must be placed in a MODE where the requirement does not apply. This is performed by placing the unit in MODE 3 within 6 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. In Mode 3, the hydrogen igniters are no longer required OPERABLE to protect the integrity of containment.

SURVEILLANCE REQUIREMENTS

SR 3.6.14.1

This SR confirms that [32 of 34] hydrogen igniters in each train can be successfully energized and assures that at least one of each redundant pair of igniters is OPERABLE in each containment region. The igniters are simple resistance elements, therefore energizing provides assurance of OPERABILITY. The allowance of up to [4] inoperable hydrogen igniters is acceptable because, although one inoperable hydrogen igniter in a region would compromise redundancy in that region, the containment regions are interconnected so that ignition in one region would cause burning to progress to the others, i.e., there is overlap in each hydrogen igniter's effectiveness between regions. Limiting this requirement to [4] inoperable hydrogen igniters is based on engineering judgment. However, per the NOTE, no 2 inoperable hydrogen igniters may exist in the same containment region. The frequency of 92 days is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.14.2

This SR requires energizing each hydrogen igniter and confirming that each is capable of producing a temperature sufficiently high to ignite a hydrogen-air mixture at the lower flammability concentration. The frequency of 18 months was based on engineering judgment, and has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors", January 1, 1988.

2. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.

3. T. E. Murley to W. S. Wilgus, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Group's Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications", May 9, 1988.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.15 Containment Recirculation Drains

BASES

BACKGROUND

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains.

The ice condenser is partitioned into 24 bays, each having a pair of inlet doors that open from the bottom plenum to allow the hot steam/air mixture from a Design Basis Accident (DBA) to enter the ice condenser. Twenty of the 24 bays have an ice condenser floor drain at the bottom to drain the melted ice water down into the lower compartment (in the 4 bays that do not have drains, the melted ice water drains through the floor drains in the adjacent bays). Each drain leads to a drain pipe that drops down a few feet, then makes one or more 90 degree bends and exits into the lower compartment. Each pipe has a gate (flapper) valve at the end which serves to keep warm air from entering during normal operation, but opens when the melted ice water exerts pressure, to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to lower containment serves to cool the atmosphere as it falls through to the floor and to provide a source of borated water at the containment sump for long term use by the Emergency Core Cooling System (ECCS), and the Containment Spray System (CSS), during the recirculation mode of operation.

The 2 refueling canal drains are at low points in the refueling canal. During a refueling, plugs are installed in the drains, and the canal is flooded to facilitate the refueling process. The water acts to shield and cool the spent fuel as it is manipulated from the reactor vessel to storage. After refueling, the canal is drained and the plugs removed. During a DBA the refueling canal drains are the main return path to the lower compartment for CSS water sprayed into the upper compartment.

The ice condenser drains and the refueling canal drains protect the integrity of the containment by functioning with the ice bed, the CSS, and the ECCS to limit the pressure and temperature that could be expected following a DBA. Maintaining containment OPERABILITY limits leakage of fission product radioactivity from containment to the environment. Loss of containment OPERABILITY could cause SITE BOUNDARY doses, in the event of a DBA, to exceed values given in 10 CFR 100 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment recirculation drains protect the integrity of the containment by functioning with the ice condenser, the CSS, and the ECCS to limit the pressure and temperature that could be expected following a DBA. The limiting DBAs considered relative to CONTAINMENT INTEGRITY are the Loss of Coolant Accident (LOCA) and the Steam Line Break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two LOCAs or SLBs are assumed to occur simultaneously or consecutively. Although the ice condenser is a passive system that requires no electrical power to perform its function, the CSS and the ARFS also function to assist the ice bed in limiting pressures and temperatures. Therefore, the postulated DBAs are analyzed, [in regards to containment Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, resulting in one train of CSS, RHR spray, EGTS, and ARFS to be inoperable].

The limiting DBA analyses (Ref. 2) show that the maximum peak containment pressure results from the [LOCA] analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and [is calculated to be less than or equal to the containment design temperature.] Therefore it is concluded that the calculated transient containment atmosphere temperatures are acceptable for the limiting DBA SLB.

In addition to calculating the overall peak containment pressures, the DBA analyses include calculation of the transient differential pressures that occur across subcompartment walls during the initial blowdown phase of the accident transient. The internal containment walls and structures are designed to withstand these local transient pressure differentials for the limiting DBAs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment recirculation drains are components that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, they satisfy the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

This LCO establishes the minimum requirements to assure that the containment recirculation drains perform their safety functions. The ice condenser floor drain valve gates must be closed to minimize air leakage into and out of the ice condenser during normal operation and must open following a DBA when melted ice water begins to drain out. The refueling canal drains must have their plugs removed and remain clear to assure the return of CSS water to lower containment in the event of a DBA. The containment recirculation drains protect the integrity of the containment by functioning with the ice condenser, ECCS, and CSS to limit the pressure and temperature that could be expected following a DBA.

APPLICABILITY

LCO 3.6.15 requires the ice condenser floor drains and the refueling canal drains OPERABLE to function with the ice condenser, ECCS, and CSS to protect the integrity of the containment by limiting the pressure and temperature that could be experienced following a DBA.

In MODES 1, 2, 3, and 4 a DBA could cause an increase in containment pressure and temperature requiring the operation of the containment recirculation drains. The possibility and consequences of these events in MODES 5 and 6 are low due to the RCS pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required OPERABLE in these MODES to protect containment OPERABILITY.

(continued)

ACTIONS

A.1

If one or more ice condenser floor drains is inoperable, the unit is not prepared to respond to the design basis events for which the drains are required. Twelve hours per drain is allowed to restore the drains to OPERABLE status before action must be taken to reduce power. The specified Completion Time is sufficient to allow the correction of minor problems or prepare the unit for an orderly shutdown and entry into Required Actions C.1 and C.2. As such, the Completion Time of 12 hours is based on engineering judgment.

B.1

If one or more refueling canal drains is inoperable, the unit is not fully prepared to respond to the design basis events for which the drains are required. A completion time of 12 hours is allowed to restore the drains to OPERABLE status before action must be taken to reduce power. The specified Completion Time is sufficient to allow the correction of minor problems or prepare the unit for an orderly shutdown and entry into Required Actions C.1 and C.2. As such, the Completion Time of 12 hours is based on engineering judgment.

C.1 and C.2

If the Required Actions are not met within the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators.

Similarly, the 30 hours allowed for reaching MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train. In Mode 5 the recirculation drains are no longer required OPERABLE to protect the integrity of containment due to the pressure and temperature limitations of this MODE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

Verifying the OPERABILITY of the refueling canal drains provides assurance that they will be able to perform their functions in the event of a DBA. This surveillance confirms that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drain, attention must be given to any debris that is located where it could be moved to the drain in the event that CSS is in operation with water flowing to the drain. SR 3.6.15.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to assure that the plugs have been removed and that no debris that could impair the drain was deposited during the time the canal was filled. The frequency of 92 days is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

SR 3.6.15.2

Verifying the OPERABILITY of the ice condenser floor drains provides assurance that they will be able to perform their functions in the event of a DBA. The inspection of the drain valve gate assures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed assures their readiness to drain water from the ice condenser. The frequency of 18 months is based on engineering judgment, and has been shown to be acceptable through industry operating experience. Because of high radiation in the vicinity of the drains during power operation, this surveillance is normally done during a shutdown.

REFERENCES

1. Title 10 of the Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone and Population Center Distance", January 1, 1988.
2. Watts Bar FSAR, Section [6].
3. NRC Interim Policy Statement, 52FR3788, "Technical Specification Improvements for Nuclear Power Reactors", February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves

BASES

BACKGROUND

The Main Steam Safety Valves (MSSVs) remove the engineered safeguards design rated steam flow while limiting the maximum steam system pressure to less than 110% of the steam generator shell side design pressure. The MSSVs pass rated flow at [3%] above the actuation pressure (Ref. 1).

Five MSSVs are provided for each steam generator in the main steam line upstream of the main steam line isolation valves. The MSSVs are self-contained, spring-loaded, and self-actuated.

APPLICABLE SAFETY ANALYSES

The MSSVs are sized to mitigate the consequences of the following transients (Refs. 1 and 2):

a. A loss of external electrical load and/or turbine trip,

b. A loss of normal feedwater,

c. A loss of offsite power to station auxiliaries (station blackout), and

d. A feedline break.

Operation of the MSSVs limits the pressure to 110% of design, thus protecting the integrity of the steam generators and secondary components from overpressure conditions. Operation of the MSSVs also indirectly limits Reactor Coolant System (RCS) temperature. Limiting RCS temperature helps the Reactor Trip System to maintain departure from nucleate boiling ratio within limits.

The steam generator and secondary components are designed to withstand loadings per the ASME Boiler and Pressure Vessel Code, Section III (Ref. 3). Limiting loadings to design values maintains the integrity of the components. These loadings are in excess of those predicted to result from the transient evaluated in the FSAR.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The MSSV is a component that is part of the primary success path and which functions or actuates to 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. As such, they satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

LCOs

Four MSSVs for each steam generator have sufficient capacity to provide overpressure protection for design basis transients occurring at 100% of RATED THERMAL POWER (RTP) (Ref. 1). However, to increase the likelihood of successful mitigation of overpressure events, five MSSVs for each steam generator are required to be OPERABLE. Operation with less than five (all) MSSVs OPERABLE for each steam generator is permissible if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER such that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20% which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity consistent with the relief capacity of the affected steam generator.

Two MSSVs have sufficient capacity to provide overpressure protection for steam generators and connected secondary components for operation in MODE 1 up to 43% of RATED THERMAL POWER (RTP) and in MODES 2 and 3 provided the Source Range Neutron Flux; Intermediate Range Neutron Flux; and/or Power Range Neutron Flux--Low trip setpoints are set per LCO 3.3.1, Reactor Trip System Instrumentation. Limiting operation to within the low power reactor trip setpoints limits THERMAL POWER to within the relief capacity of the OPERABLE MSSVs.

Below 10% of RTP (P-10 Setpoint), the Power Range Neutron flux--Low setpoint trip function is enabled when the appropriate number of power range channels are below the P-10 setpoint.

(continued)

BASES (continued)

LCOs (continued)

For each steam generator, at a specified pressure, the Fractional Relief Capacity, FRCi, of each MSSV is determined as follows

$$FRCi = \frac{A}{B}$$

Where A is the relief capacity of the MSSV, and
B is the total relief capacity of all the MSSVs of the steam generator.

These fractional relief capacities are used to establish the Power Range Neutron Flux--High trip setpoints for operation with less than five MSSVs OPERABLE for each steam generator.

The Power Range Neutron Flux--High trip setpoints specified in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. These trip setpoints are determined as follows

SP = [1 - (N1*FRC1 + N2*FRC2 + ...+ N5*FRC5)] * RP

Where

SP is the Power Range Neutron Flux--High trip setpoints calculated for LCO 3.7.1 for operation with less than five MSSVs OPERABLE

N1, N2,..,N5 is the status of the MSSV 1, 2,..,5
= 0 if the MSSV is OPERABLE
= 1 if the MSSV is inoperable

FRC1,FRC2,...,FRC5 is the relief capacity of the MSSV 1, 2, ..5 as defined above

RP is the Power Range Neutron Flux--High trip setpoints as specified in Table 3.3.1-1, function 2.A of LCO 3.3.1, Reactor Trip System Instrumentation.

(continued)

BASES (continued)

APPLICABILITY

Design bases transients occurring during MODE 1 are predicted to result in lifting of the MSSVs (Refs. 1 and 2).

Nominal secondary side operating pressure is highest at low power and no-load temperature conditions. To prevent overpressurization of steam generators as a result of a transient, MSSV OPERABILITY is also required during MODES 2 and 3.

Operation in MODES 4 and 5 does not require MSSV OPERABILITY since, in these MODES, it is improbable that an overpressure condition can be achieved due to the reduced nominal operating pressure and temperature and the lack of significant THERMAL POWER.

ACTIONS

A.1 and A.2.1

Five MSSVs are required to be OPERABLE for each steam generator. However, four MSSVs for each steam generator have sufficient capacity to provide overpressure protection for design basis transients occurring at 100% of RTP (Ref. 1). In recognition of having this extra relief capacity at 100% of RTP and to maintain this extra relief capacity in the event that one through three MSSVs in one or more steam generators becomes inoperable, Required Action A.1 allows 4 hours to determine the appropriate number of MSSVs for each steam generator to be OPERABLE. The Completion Time of 4 hours is based on engineering judgment. The required number of MSSVs are to be returned to OPERABLE in this same 4 hours or further measures must be taken.

A.2.2.1

Within the same Completion Time of 4 hours, Action A.2.2.1 requires that THERMAL POWER be reduced to a level consistent with the reduced relief capacity. The Completion Time of 4 hours is based on engineering judgment considering the consequences of failure of one through three MSSVs in one or more steam generators.

A.2.2.2

Subsequent Action A.2.2.2 can be completed without challenging the Reactor Trip System. An additional time of 4 hours is allowed to reduce the Power Range Neutron Flux--High trip

(continued)

BASES (continued)

ACTIONS (continued)

A.2.2.2 (continued)

setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of 4 hours. The Completion Time of 8 hours is based on operating unit experience in resetting all channels of a protective function and on the improbability of the occurence of a transient which could result in steam generator overpressure.

Completion of Action A.1 and A.2.1 or Actions A.2.2.1 and A.2.2.2 results in a relief capacity consistent with THERMAL POWER.

B.1 and B.2

If the one through three inoperable MSSVs are not restored to OPERABLE status within a Completion Time of 4 hours or the Power Range Neutron Flux--High trip setpoints not reset within a Completion Time of 8 hours, Required Action B.1 directs the unit to a MODE in which more than one MSSV may be inoperable without consideration for the Power Range Neutron Flux--High trip setpoints. Required Action B.1 directs the unit to MODE 3 in 6 hours and MODE 4 within 12.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

MSSV lift settings are established per the ASME Boiler and Pressure Vessel Code, Section III (Ref. 3) as recorded in the FSAR (Ref. 2) and WCAP-7769 (Ref. 1). The MSSV lift setting tolerance of ± [1%] and verification of MSSV lift setting is per [ASME Boiler and Pressure Vessel Code, Section III and XI as supplemented by ANSI/ASME OM-1-1981] (Ref. 5). Reference 5 requires that testing be performed using saturated steam with the valve at nominal operating temperature unless a correlation has been established allowing testing at other conditions. If tested at conditions other than nominal operating temperature with saturated steam, the lift setting at which the valve is set during the test shall equate to the lift setting at nominal operating temperature with saturated steam. That is, when

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.1 (continued)

the valve is taken to nominal operating conditions, any shift in the setpoint from the set during the test will result in the setpoint being at the setpoint for saturated steam at nominal operating temperature conditions. Surveillance Requirements are specified in the Inservice Inspection and Testing Program. Section XI of the ASME code provides the activities and their frequencies necessary to satisfy the requirements. No additional requirements are specified.

One method of testing the MSSVs is to test in place which may require the unit to be at nominal operating pressure and temperature. An exception to SR 3.0.4 is required to allow the unit to be taken to nominal operating pressure and temperature. The exception is reasonable based on engineering judgment considering the time required to verify MSSVs on each steam generator OPERABLE; the low probability that the MSSVs, prior to lift setting verification, would not perform the intended function if required; and the low probability that an overpressure event would occur during this time period.

(continued)

BASES (continued)

REFERENCES

1. Cooper, K., R.M. Starck, V. Miselis, S.J. Sarver, and M.A. Mangan, "Topical Report Overpressure Protection for Westinghouse Pressurized Water Reactors", WCAP-7769 Rev.1, June 1972.

2. Watts Bar FSAR Section 15.2 Condition II - Faults of Moderate Frequency, Section 15.4 Condition IV - Limiting Faults, and Section 10.3 Main Steam Supply System.

3. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, and Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers, New York, [1971 thru Summer 1973 Addenda].

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

5. [ANSI/ASME OM-1-1981, "Requirements for In-service Performance Testing of Nuclear Power Plant Pressure Relief Devices".]

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Line Isolation Valves (MSIVs)

BASES

BACKGROUND

The OPERABILITY of the MSIVs and associated bypasses ensures that no more than one steam generator will blow down in the event of a Main Steam Line Break (MSLB). This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System (RCS) cooldown associated with the blowdown, and (2) limit the pressure rise within the containment in the event the MSLB occurs within the containment. The MSIVs also isolate the ruptured steam generator after a Steam Generator Tube Rupture (SGTR).

One MSIV and associated bypass valve are installed in each steam generator steam line outside containment, downstream of the main steam safety valves. The MSIVs are designed to close automatically after receipt of a [High Steam Flow with Lo-Lo T_{avg} or Lo Steam Line Pressure; or Containment Pressure--High High] signal for closure. The MSIVs fail closed on loss of actuation capability. Each MSIV has a MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs.

With one MSIV inoperable in the open position, only one steam generator will blow down in the event of a MSLB. This assumes no additional failures.

The MSIVs serve an isolation safety function only and remain open during power operation. They are required to limit uncontrolled flow of steam from the steam generators in the event of a MSLB. These valves operate under the following situations:

a. A MSLB inside the containment.

If the break is within the containment, steam is discharged into the containment. The other steam generators act to feed steam through the interconnecting header downstream of the MSIVS and back to the broken line and then into the containment. Mass and energy release from a MSLB results in a significant pressure and temperature increases in the containment. Reverse flow protection is necessary to prevent discharge of more than one steam generator. According

(continued)

BASES (continued)

BACKGROUND (continued)

to calculations, reverse flow must be interrupted to limit the containment pressure rise to an amount below the design pressure. Closure of these valves allows for a single failure of an active component.

b. A MSLB outside containment and upstream from the MSIVs is not a containment pressurization concern. However, the uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition.

c. A MSLB downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. After a SGTR, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators. In addition to minimizing radiological releases, this enables the operator to establish a pressure difference between the ruptured and intact steam generators, as a necessary step toward terminating the flow through the rupture.

The MSIVs are required by 10 CFR 50, Appendix A, General Design Criteria (GDC) 57 (Ref. 1a) to isolate the main steam line penetrating containment, and by GDC 38 (Ref. 1b) to ensure that the consequences of the Design Basis events do not exceed the capacity of the containment heat removal systems.

APPLICABLE SAFETY ANALYSES

The steam release arising from a MSLB would result in an initial increase in steam flow and then a decrease in steam flow as the steam pressure falls. Energy removal from the RCS reduces the coolant temperature and pressure. When the moderator temperature coefficient is negative, the cooldown causes an insertion of positive reactivity, reducing SHUTDOWN MARGIN. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a MSLB is a potential problem because of the high power peaking factors that exist as a result of the stuck rod assumption and non-uniform core inlet temperatures. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The MSIV design basis is established by the large MSLB inside containment (Ref. 2). The design assumptions include the blowdown of no more than one steam generator, assuming the failure of one MSIV to close on demand.

The safety analysis evaluates various MSLBs using appropriate acceptance criteria. The large MSLB outside containment, upstream of the MSIV, is limiting for offsite dose, even though a break in such a short length of main steam piping is of a very low probability. The large MSLB inside containment at hot zero power is the limiting case for a post-trip return to power. The events are analyzed both with and without offsite power available. With offsite power available, the reactor coolant pumps continue to circulate primary coolant through the steam generators which maximizes RCS cooldown. Without offsite power available, the response of mitigating systems are delayed. Significant single failures considered include failure of a MSIV to close. The MSIVs are also used during both feedwater line break and SGTR events.

The MSIV is a component that is part of the primary success path and which functions or actuates to 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. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

LCOs

The MSIVs must be OPERABLE whenever significant mass and energy are present in the RCS and steam generators. This assures that in the event of a MSLB, a single failure cannot result in the blowdown of more than one steam generator.

APPLICABILITY

The MSIVs must be OPERABLE whenever the steam generators could be required for primary heat removal. This includes MODES 1, 2, and 3. The MSIVs typically remain closed during heatup and are not opened until ready for surveillance testing. In MODE 2 or 3 the applicability for OPERABLE valves refers to the open MSIV or bypasses.

(continued)

BASES (continued)

APPLICABILITY (continued)

The MSIVs need not be OPERABLE when in MODES 4, 5, and 6 because a MSLB with subsequent blowdown of more than one steam generator is not considered a credible accident.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, time is allowed for repair. The Completion Time of 8 hours for repair is based on engineering judgment and the low probability of an event occuring within this time period which requires MSIV actuation.

B.1

Since the MSIVs are required to be OPERABLE in MODE 1, the inoperable MSIVs must be closed. If closed, the affected MSIVs are placed in the position required for performing their safety function as assumed by the safety analysis. The Completion Time of 6 hours is sufficient to conduct a controlled shutdown to a point where the inoperable MSIVs may be closed and is based upon industry operating experience and engineering judgment.

C.1 and C.2

If the MSIV cannot be closed, a Completion Time of 6 hours is sufficient to reach MODE 3 and another 6 hours to reach MODE 4. This Completion Time of 6 hours is based upon industry operating experience, without challenging either unit systems or operating personnel. This places the unit in a MODE in which MSIVs OPERABILITY is not required.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

The MSIV closure time is assumed in the safety analyses. The surveillance is normally performed in conjunction with a refueling outage. Surveillance Requirements and their frequencies are specified in the Inservice and Inspection Testing Program.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.1 (continued)

The amount of steam flow in MODE 4 or below is significantly lower than it is in MODE 3 or above. The MSIVs are designed to operate in conditions when the steam flow is greater than that typically available in MODE 4. Operating experience demonstrates that closure times measured in MODE 4 or below are often larger than required, and that closure times are shortened by having increased steam flow. The Note allows for delaying testing to MODE 3 in order to obtain closure times for conditions consistent with the conditions under which the acceptance criterion was generated.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants:

   a. Criterion 57 - Closed System Isolation Valves.
   b. Criterion 38 - Containment Heat Removal.

2. Watts Bar FSAR Section [6.2], Containment Systems.

3. ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, Article IWV-3400 "In-service Tests - Category A and B Valves", American Society of Mechanical Engineers, New York, [1971 thru Summer 1973 addenda].

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulation and Isolation Valves (MFRVs and MFIVs)

BASES

BACKGROUND

The MFRVs and the MFIVs are in series in the Main Feedwater (MFW) line upstream of each steam generator. The bypass MRFV and bypass MFIV are located in the bypass line to the upper nozzle of each steam generator. These valves provide the two valve isolation of each MFW line.

The principle function of the MFIVs and bypass MFIVs is to isolate MFW flow to the secondary side of the steam generators following a high energy line break. The safety-related function of the MFRVs and the MFRV bypass valves, as covered by this LCO, is to provide the second isolation for MFW flow to the secondary side of the steam generators following a high energy line break. Closure of all these valves terminates feedwater flow to the steam generators, preventing uncontrolled blowdown from the steam generator in the event of a feedwater line break occurring upstream of the valves. Consequences of events occurring in the main steam lines or MFW lines downstream of the valves will also be mitigated by closing the valves. Closure of the valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks and minimizing the positive reactivity effects of the Reactor Coolant System (RCS) cooldown associated with the blowdown.

The MFIVs and bypass MFIVs isolate the nonsafety-related portions from the safety-related portion of the system. In the event of a secondary cycle pipe rupture inside the containment, the valves limit the quantity of high energy fluid that enters the containment through the broken loop and provides a pressure boundary for the controlled addition of Auxiliary Feedwater (AFW) to the three intact loops. The MFW check valve in each loop provides backup isolation.

(continued)

BASES (continued)

BACKGROUND (continued)

The MFIV is located in each MFW line and the bypass MFIV is located in each upper nozzle feedwater line outside but close to containment. [The bypass MFIVs are located upstream of the AFW System injection point so that AFW may be supplied to the steam generators following valve closure.] The piping volume from the MFIVs to the steam generators must be accounted for in calculating mass and energy releases [and must be refilled prior to AFW reaching the steam generators] following either a steam or feedwater line break.

The valves close on receipt of [Safety Injection, Steam Generator Water Level--High-High, and Reactor Trip with Lo-T_{avg}] signal. The valves may also be actuated manually.

A description of the valves is found in the FSAR (Ref. 1). The [MFIVs and a check valve] are required by 10 CFR 50, Appendix A, General Design Criteria (GDC) 57, (Ref. 2) to isolate the feedwater line penetrating containment, and by GDC 38 (Ref. 3) to ensure the consequences of events do not exceed the capacity of the containment heat removal systems.

APPLICABLE SAFETY ANALYSES

The design basis of the valves is established by the large steam line break. It is also influenced by large feedwater line breaks occurring upstream of the MFIV or inside containment. [Closure of the valves is also relied on to terminate an excess feedwater event upon receipt of a Steam Generator Water Level--High-High, P-14, signal.]

Failure of the valves to close, following a steam line break, feedwater line break, or excess feedwater event, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following a steam line break or feedwater line break event.

These valves are components that are part of the primary success path and which function or actuate to 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. As such, the valves satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

(continued)

BASES (continued)

LCOs

Following a feedwater or main steam line break, this LCO ensures that the MFIVs and MFRVs will isolate MFW flow to the steam generators. The MFIVs and bypass MFIVs also isolate the nonsafety-related portions from the safety-related portions of the system.

Failure to meet the LCO requirements could result in additional mass and energy being released following a steam or feedwater line break. [With Steam Generator Water Level--High-High, P-14, relied on to terminate an excess feedwater flow event and provide equipment protection, failure to meet the LCO could result in the introduction of water into the main steam lines.]

APPLICABILITY

The MFRVs and MFIVs must be OPERABLE whenever MFW is being supplied to the steam generators. This is generally the case in MODES [1, 2, and 3]. The MFRVs and MFIVs shall be OPERABLE or closed to isolate a fault in the MFW piping, should one occur, and ensure AFW to the steam generators.

ACTIONS

A.1 and A.2

With one [unisolated] MFRV or MFIV in one or more flow paths inoperable during MODES 1, 2, and 3, the Required Action is to restore the valve to OPERABLE status or to close or isolate the valve. [As (three) loop operation is not permitted,] the unit must ultimately be placed in MODE 3 with the valve closed [or isolated] if the valve is not restored to OPERABLE status. It may be a unit operating constraint that prior to isolating the feedwater flow to an inoperable MFRV or MFIV in the main line, both the feedwater flow and THERMAL POWER should be reduced. This reduction is determined by unit experience operating at THERMAL POWER with feedwater flow through the bypass MFRV valve. The intent is to establish an alternate feedwater flow providing adequate steam generator feed through the bypass MFRV valve. The MFRV or MFIV then may be isolated minimizing unit exposure to Steam Generator Water Level--Low trip.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2 (continued)

The Completion Time of 72 hours is comparable to that for other ESF components. This is acceptable due to the low probability of the events requiring the MFRVs and MFIVs and the availability of backups by the remaining inline MFIV or MFRV to terminate MFW flow.

B.1 and B.2

With one MFRV and MFIV inoperable in the same flow path, operations may continue provided at least one MFRV or MFIV is restored to OPERABLE status or the inoperable MFRV or MFIV is closed or isolated. A Completion Time of 8 hours for this condition exists to provide time for repair and to ensure that the MFRVs or MFIVs are closed, or isolated, to isolate a fault in the MFW piping should one occur if the valve cannot be made OPERABLE.

C.1 and C.2

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. A Completion Time of 6 hours for reaching MODE 3 and reaching MODE 4 in the following 6 hours is a reasonable time, based on industry operating experience without challenging safety systems or operators. This action minimizes the risk by placing the unit in a MODE in which the LCO is not applicable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

The response time of the MFRVs and MFIVs is assumed in the accident and containment analyses. Surveillance Requirements are specified in the Inservice Inspection and Testing Program. No additional requirements are specified. SR 3.0.4 is not applicable to this Surveillance Requirement for entry into MODE 3.

REFERENCES

1. Watts Bar FSAR, Section [10.4.7], Condensate and Feedwater System.
2. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants: Criterion 57, Closed System Isolation Valves.
3. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants: Criterion 38, Containment Heat Removal.
4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater System

BASES

BACKGROUND

The main purpose of the Auxiliary Feedwater (AFW) System is to provide feedwater to the steam generators following a loss of normal feedwater. This AFW is supplied from the Condensate Storage Tank (CST) to the steam generator secondary side using the AFW pumps. The steam generators function as a heat sink for core decay heat loads. The heat load is dissipated by either venting to the condenser through the steam dump valves or to atmosphere through the atmospheric relief valves, or steam generator safety valves. The AFW System has three auxiliary feedwater pumps and redundant flow paths for adequate flow following any single active failure.

The AFW System includes two motor-driven pumps and one turbine-driven pump. Each motor-driven pump provides 100% of the required capacity to the steam generators as assumed in the safety analysis. The turbine-driven pump has twice that capacity. The 100% capacity will remove decay heat and cool the unit to MODE 4 . Each motor-driven pump is powered from a separate Engineered Safety Feature electrical bus. The turbine-driven pump has steam supplies from two main steam lines upstream of the main steam isolation valve. Each steam feed will supply 100% of the turbine-driven AFW pump requirements. Having steam and electric motor-driven AFW pumps meets the requirements of Branch Technical Position ASB 10-1.

The AFW System is required to meet both 10CFR50, Appendix A, General Design Criteria (GDC) 34 (Rev. 1a) for residual heat removal and GDC 44 (Rev. 1b) for cooling water needs. These criteria identify the capability to both remove decay heat and other residual heat and transfer it to the heat sink.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event that causes a loss of normal feedwater. The design basis of the AFW system is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus [2%] accumulation. In addition, the AFW system must supply enough makeup water to replace steam generator secondary inventory being lost as steam as the unit cools to MODE 4 conditions. Sufficient AFW flow must be available to account for flow losses such as pump recirculation and line breaks.

The following are the limiting Design Basis Accidents (DBAs) for this system:

a. Feedwater System Pipe Rupture, and

b. Loss of Normal Feedwater

As the main feedwater pumps are isolated on a unit trip, the AFW system is activated for all DBAs including reactor trip.

If adequate feedwater is not maintained in the steam generators, the Reactor Coolant System (RCS) temperature and pressure will rise and threaten the integrity of both the fuel cladding and the reactor coolant pressure boundary. These are the safety limits protected by the AFW System.

Analytical criteria have been defined to help demonstrate that the safety limits are not violated. These criteria are:

a. Pressure in both the Reactor Coolant and Main Steam Systems shall be maintained below 110% of the design pressure (Ref. 2),

b. Fuel cladding integrity shall be maintained by ensuring that the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the 95/95 DNBR limits for Condition 2 events, and

c. The core remains in place and geometrically intact with no loss of core cooling capability for Condition 4 events

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The AFW System is a part of the primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

Each motor-driven AFW Pump feeds 2 steam generators through individual air-operated flow control valves. The turbine-driven pump feeds all 4 steam generators through individual air-operated flow control valves. With these conditions, three AFW pumps are required to be OPERABLE so that AFW will be available for heat removal for all events accompanied by a loss of offsite power and a single failure. The analysis assumes that all AFW to the faulted steam generator is lost. An additional steam generator is assumed to lose all AFW flow as a result of a single failure. If less than the required components of the AFW System are OPERABLE, the single active failure criterion cannot be met.

APPLICABILITY

AFW System MODE applicability is based on design requirements that both the AFW System and steam generators have the capability of cooling the unit to the temperature and pressure where the Residual Heat Removal (RHR) System can assume the heat load.

The AFW System must be OPERABLE in MODES 1 and 2 so that it will be available after a DBA which results in a reactor trip. In MODE 3, the AFW System cools the RCS, directing the unit to enter MODE 4 where the RHR System is placed in operation. MODE 4 is applicable if the steam generator is relied upon for heat removal.

ACTIONS

A.1

With one steam supply to the turbine-driven AFW pump inoperable, a Completion Time of 7 days is allowed to return the steam supply to OPERABLE status.

B.1

With one AFW train inoperable for reasons other than condition A, 72 hours are allowed to return an AFW train to OPERABLE status.

(continued)

BASES (continued)

ACTIONS (continued)

C.1

With two AFW subsystems inoperable, or the Required Actions and associated Completion Times for Condition A or B are not met, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 4 in the next 6 hours.

D.1

With no AFW subsystems OPERABLE, the only action is to restore a subsystem as soon as possible. Should the unit be shut down under this condition, there would be no makeup water to the steam generators to cool the unit to MODE 4 conditions, where the RHR System would continue to cool the RCS. Action should be initiated immediately to restore one train to OPERABLE status and continue as required.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

During normal unit operations, the AFW System is not operating but is aligned for standby operation. In this condition, the manual valves should all be positioned so that the only action needed to begin AFW operation is for the pumps to start. The power-operated and automatic valves (excluding check valves) should be correctly positioned for the existing system conditions. The surveillance Frequency of 31 days is based on engineering judgment. It takes into account the importance of the valves, the probability of being left in the wrong position, and their location and physical environment.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Periodic surveillance testing of the AFW pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of ASME Code. This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump characteristic curve. This verifies that both 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 unit safety analysis. The Surveillance Requirements, specified in the Inservice and Inspection Testing Program, provide the activities necessary to satisfy the requirements. The 31 day frequency is based on engineering judgement.

SR 3.7.5.3

AFW System automatic valves (excluding check valves), if not in the proper position, will be actuated by the AFW Actuation Signal. The AFW System actuates on a [Safety Injection, Loss of Offsite Power, and Steam Generator Level--Low Low] signal. This surveillance is performed to ensure the AFW System automatic valves will properly actuate on receipt of an AFW Actuation Signal. The Surveillance Frequency of 18 months for the testing of the AFW System response is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.5.4

Confirmation that each AFW pump starts on receipt of the AFW Actuation Signal is required every 18 months. This surveillance is performed to ensure the AFW pumps will properly start on receipt of the AFW Actuation Signal. The surveillance Frequency of 18 months is based upon engineering judgment and has been shown to be acceptable through operating industry experience.

SR 3.7.5.5

The intent is to verify the AFW System flowpaths, prior to entering MODE 2 operation, after > 30 days in MODE 5. This assures the AFW System is properly aligned after an extended unit outage. Operability of the AFW flowpaths must be demonstrated before sufficient core heat is generated requiring operation of the AFW System during subsequent unit shutdown.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants:

   a. Criterion 34 - Residual Heat Removal.
   b. Criterion 44 - Cooling Water.

2. Watts Bar FSAR, Section [15.2.8], Loss of Normal Feedwater.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST is a covered tank. It serves as the main source of water to the Auxiliary Feedwater (AFW) System pumps. During unit cooldown operations, upon AFW low suction pressure, the AFW water supply is transferred to the ERCW System.

The principle function of the CST is to provide a source of water to the steam generators for removal of decay and sensible heat from the Reactor Coolant System (RCS). The CST provides water to the AFW pumps. The AFW pumps supply the water to the steam generators where it removes heat from the RCS. The generated steam is then released by the main steam safety valves, condenser dump valves, or the atmospheric dump valves.

The water volume of the CST allows the unit to be maintained in MODE 3 for a minimum of [2] hours and then allows a subsequent cooldown to the Residual Heat Removal (RHR) System initiation condition. The volume of water contained by the CST is greater than the volume of water required to be delivered to the AFW.

The CST is required to meet 10 CFR 50, Appendix A, General Design Criteria (GDC) 34 (Ref. 1a) and GDC 44 (Ref. 1b) regarding the capability to remove decay heat.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and cooldown the unit, following all events in the accident analyses except for the large break loss of coolant accident. The limiting event associated with CST volume is a large feedwater line break with loss of offsite power. Other single failures that affect this event include the following:

a. The failure of the diesel generator powering the motor-driven AFW pump which supplies the unaffected steam generators, and

b. The failure of the steam turbine-driven AFW pump.

A non-limiting event associated with CST inventory determination is a break in a main feedwater line or AFW line near the common header. This has the potential of causing an uncontrolled inventory loss until terminated by operator action. This condensate inventory loss is partially compensated for by the steam generator inventory retention in this event due to the presence of check-valves in the main feedwater lines.

The CST volume is a process variable that is an initial condition of a Design Basis Accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

In addition, the CST is a component that is a part of the primary success path and which functions or actuates to 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. As such, it satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 2).

(continued)

BASES (continued)

LCOs

The CST must contain sufficient cooling water to remove decay heat following both a steam line break and a reactor trip and then cooldown the RCS to RHR initiation conditions, assuming a loss of offsite power and the most adverse single failure. During this event, sufficient water inventory must be maintained to ensure adequate net positive suction head for the AFW pumps, as well as to account for any losses from either the turbine-driven AFW pump or through pipe breaks.

The volume specified is based on holding the unit in MODE 3 for [2] hours with no steam line break followed by cooldown to RHR entry conditions. This basis is established by BTP RSB 5-1 (Ref. 3) and exceeds the volume required by the accident analyses.

APPLICABILITY

The required CST water volume must be available whenever the AFW is required to be OPERABLE to provide a heat sink for unit heat removal. Since the AFW is required to be OPERABLE in MODES 1, 2, and 3, the CST applicability is limited to the same MODES. MODE 4 is applicable if the steam generator is relied upon for heat removal.

ACTIONS

A.1

With the CST unable to supply the required volume of cooling water to the AFW pumps, it must be restored to OPERABLE status. The Completion Time of 4 hours is considered adequate to restore the required volume from the [Demineralized Water Makeup System]. This Completion Time of 4 hours is sufficient to correct minor problems or to prepare the unit for an orderly shutdown and entry into Condition B.

A.2.1 and A.2.2

The ERCW system bakcup water supply may be verified OPERABLE prior to the expiration of the 4 hours authorized for CST restoration. Required Action A.2.1 verifies that a backup water supply from ERCW is OPERABLE, allowing for more time for CST restoration. The Completion Time of 7 days is consistent with the required Completion Times for restoring operation to within LCO limits.

(continued)

BASES (continued0

ACTIONS (continued)

B.1 and B.2

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown shall be commenced. The Completion Time of 6 hours specified for reaching MODE 3 and the Completion Time of 12 hours for reaching MODE 4 without reliance on the steam generators is based on engineering judgment and industry operating experience without challenging either safety systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Verifying the CST water level ensures the required volume of cooling water. The surveillance frequency of 12 hours is adequate as the operator should be aware of any unit conditions which may affect the CST volume.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants:

   a. Criterion 34 - Residual Heat Removal.
   b. Criterion 44 - Cooling Water.

2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

3. Branch Technical Position RSB 5-1, Rev.[ ], "Design Requirement of the Residual Heat Removal System", [ ].

B 3.7 PLANT SYSTEMS

B 3.7.7 Secondary Coolant Specific Activity

BASES

BACKGROUND

The secondary coolant systems are closed thermodynamic systems used to maintain the water level in the steam generators for the purpose of cooling the primary system and steam generation. The secondary coolant systems include the feedwater pumps, steam generator shell side, steam piping to the plant turbines, and the drain and condensate systems which return water to the feedwater pumps for recirculation.

The secondary coolant specific activity is concerned with the radioactivity conditions of the feedwater in the shell side of the steam generators. The presence of radioactivity in the secondary coolant systems is primarily the result of reactor coolant leakage through the steam generator tubes to the feedwater in the steam generator shell side. The feedwater is drawn off from the shell side of the steam generator through the steam generator blowdown lines to the sampling system for both chemical and radiochemical analyses. [The sample connection from the blowdown lines is located as close to the steam generator as possible to minimize the transit time of the steam generator water mass to the sampling point to ensure representative sampling.]

The specific activity of a secondary coolant sample is determined by a gamma isotopic analysis. This analysis is a quantitative measurement of total specific activity for radionuclides with half-lives > 10 minutes based upon energy peaks identifiable with 95% confidence levels. The DOSE EQUIVALENT I-131 is taken from this measurement. While the other isotopes are not part of this LCO, they would be available for trending if necessary.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

While in MODES 1, 2, 3, or 4, secondary coolant specific activity of [0.1] μCi/g DOSE EQUIVALENT I-131 is an initial condition parameter in the analysis (Ref. 1) of the postulated Main Steam Line Break (MSLB) accident. Other initial condition parameters used in the analysis are reactor coolant specific activity of [1.0] μCi/g DOSE EQUIVALENT I-131 with large iodine spike values persisting for the duration of the accident, the most reactive rod control cluster assembly stuck in its fully withdrawn position, a 1 gpm steam generator primary-to-secondary leak to all steam generators, and concurrent loss of offsite power. The accident analysis also includes a case for reactor coolant activity spiking to 60 μCi/g DOSE EQUIVALENT I-131. The steam generator connected to the ruptured steam line is assumed to boil dry. The remaining steam generators are isolated by the closure of their main steam isolation valves. These actions are initiated by the Engineered Safety Features Actuation System under the condition with [High Steam Flow with Steam Line Pressure--Low or T_{avg}--Low Low; or Containment Pressure High High], as described in the steam system piping failure accident analysis (Ref. 1).

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the safety relief valves and steam generator Atmospheric Relief Valves (ARVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the unit cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to be available to discharge steam and any entrained activity through the safety relief valves and steam generator atmospheric relief valves (ARVs) during the event. 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The accident analysis of the MSLB failure (Ref. 1) assumes the secondary coolant specific activity to have a radioactive isotope concentration of [0.1] μCi/g 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 a MSLB are well within the Exclusion Area boundary limits of 10 CFR 100 (Ref. 2) for whole body and thyroid dose rates as required by the NRC Standard Review Plan (Ref. 3).

The secondary coolant specific activity is a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 4).

LCOs

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant systems of ≤ [0.1] μCi/g DOSE EQUIVALENT I-131 is required to contain the radiological consequences of a MSLB accident within acceptable limits of 10 CFR 100 (Ref. 2) and the NRC Standard Review Plan (Ref. 3).

Monitoring of DOSE EQUIVALENT I-131 in the secondary coolant systems is necessary to ensure that appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a MSLB accident and to correct the conditions causing the secondary coolant specific activity to exceed [0.1] μCi/g DOSE EQUIVALENT I-131.

(continued)

BASES (continued)

APPLICABILITY

The secondary coolant specific activity limitation of ≤ [0.1] μCi/g DOSE EQUIVALENT I-131 contained in LCO 3.7.7 is applicable to operation in MODES 1, 2, 3, and 4. In these MODES the primary and secondary coolant systems temperature and pressure conditions are sufficient for steam generator operation and steam generation. For MODES 1, 2, and 3, the temperature conditions are such that significant steam generation is present throughout the temperature ranges defined for these MODES. In MODE 4, the primary temperature condition for steam generation is present upon entry from MODE 3 and will continue with reduced volume throughout most of MODE 4.

In MODES 5 and 6, with the primary coolant temperature ≤ 200°F, sufficient energy is not present for steam generation.

ACTIONS

A.1 and A.2

In MODES 1, 2, 3, or 4, steam generation and flow are present and sufficient to cause a release of radioactivity to the atmosphere under the postulated MSLB accident conditions. The initial action for specific activity not within limits is to be in MODE 3 and is considered to be the first step in the normal progression for an orderly change from power operation to a shutdown condition. A Completion Time of 6 hours is a reasonable time, based on engineering judgement, to reach MODE 3 from full power without challenging safety systems. In addition to the going to MODE 3, the plant must be cooled to MODE 5. In MODE 5, primary coolant temperature, T_{avg} is ≤ 200°F, steam generation is not possible, and the postulated MSLB accident conditions do not apply. The Completion Time of an additional 30 hours to reach MODE 5 is reasonable considering that a unit can easily cool down in such a time frame on one safety system train.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Performance of this surveillance with a frequency of [31] days provides monitoring for the presence of radioiodines in the secondary coolant. This frequency of iodine concentration analysis allows the level of DOSE EQUIVALENT I-131 to be monitored, increasing trends to be detected, and appropriate action to be taken to maintain levels below the LCO limit. Other instrumentation is available such as the steam generator blowdown monitors and the condenser air ejector monitors that would allow detection of a significant steam generator tube leak. The frequencies have been shown to be acceptable through industry operating experience.

REFERENCES

1. Watts Bar FSAR, Chapter 15, Accident Analysis.

2. Title 10 Code of Federal Regulations, Part 100, Reactor Site Criteria, Section 11, Determination of Exclusion Area, Low Population Zone, and Population Center Distance.

3. NUREG-0800, Standard Review Plan, Section 15.1.5, Steam System Piping Failures Inside and Outside of Containment, and Appendix A, Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR, [ ].

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements For Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.8 Component Cooling Water System

BASES

BACKGROUND

The Component Cooling Water System (CCS) provides a continuous supply of cooling water to unit components which handle potentially radioactive fluids. The CCS System is a closed system which forms an intermediate barrier between potentially radioactive systems and the Essential Raw Cooling Water System (ERCW) to reduce the possibility of releasing fission product radioactivity to the environment.

CCS must be supplied to Nuclear Steam Supply System (NSSS) components as required to accomplish the following unit functions:

a. Removal of residual and sensible heat from the Reactor Coolant System through the Residual Heat Removal (RHR) System during unit cool down,

b. Cooling of letdown flow to the Chemical and Volume Control System during power generation,

c. Cooling of Engineered Safety Features (ESF) loads after a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB), and

d. Removal of heat from various NSSS components during power generation and during unit shutdown.

The CCS System consists of two separate, 100% capacity trains. Separate Train A equipment is provided in each unit, whereas Train B is shared by both units. Train A in Unit 1 is served by CCS Hx A and CCS pump 1A-A. Pump 1B-B, which is actually Train B equipment, is also normally aligned to the Train A header in Unit 1. However, pump 1B-B can be realigned to Train B on loss of Train A.

Similarly, Train A in Unit 2 is served by CCS Hx B and CCS pump 2A-A with support from pump 2B-B.

Train B in both units is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B headers since few nonessential, normally-operating loads are assigned to Train B. However, pumps 1B-B and 2B-B can be realigned to the

(continued)

BASES (continued)

BACKGROUND

Train B headers if necessary. Pump C-S is normally powered from Train B electrical circuits. However, the electrical system has been designed so that the power feed can be switched to Train A. If Train B has failed, valves can then be repositioned to align pump C-S to either Train A header and thereby provide additional Train A capacity if needed.

Surge tanks have been provided in each unit to allow for volumetric changes in the cooling water during operation to provide a point for adding makeup water, corrosion inhibitors, etc. A baffle has been provided in each tank to separate the volume into a Train A and Train B half. Each baffle extends only part-way up the tank, so that the water is shared above that point and dedicated to a train below. This feature prevents a leak in one train from draining water from the other train below a safe volume.

The CCS is considered an ESF System because it is vital during recovery from an accident. To this end, the system is designed to meet Class I requirements for structural integrity and reliability required by 10 CFR 50.

During power generation, one CCS train is normally operating to supply one train of safety-related and the nonsafety-related cooling loads. On receipt of a Safety Injection Signal, the CCS pumps are automatically started. Although not explicitly covered by this LCO, the CCS System is used for unit cool down and during unit shutdown to provide cooling water to the RHR heat exchangers in MODES 5 and 6; spent fuel cooling may also be required.

The CCS System is designed to normally supply cooling water at a temperature of [95]°F. During initial operation of the RHR System for unit cool down, the CCS System equilibrium temperature at the CCS heat exchanger outlet may rise to [110]°F (Ref. 1).

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the CCS System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The CCS System is designed to remove heat from components important to mitigate the consequences of a LOCA or MSLB and to transfer the heat to the ERCW system. The redundant cooling capacity of this system and each of the safety-related systems which it serves, assuming

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

a single failure coincident with the loss of offsite power, is consistent with the assumptions used in the safety analysis (Ref. 2). All vital power can be supplied from either onsite or offsite power systems.

Determination of the minimum flows to each of the required cooling loads includes the assumption of the highest anticipated CCS temperature. The CCS design assures that these minimum flows to ESF equipment are achieved by the system in its accident configuration, one pump per train in operation. The CCS System is a support system to various systems which are assumed to function to mitigate various Design Basis Accidents (DBAs). The CCS is a system that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, the CCS System satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The CCS trains are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a LOCA or MSLB, one train of CCS is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CCS must be OPERABLE. At least one train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

APPLICABILITY

The CCS System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal operations and during and after a LOCA or MSLB accident. In MODES 1, 2, 3, and 4, a LOCA or MSLB could require service of these systems supported by CCS. The possibility of these events in MODES 5 and 6 is low due to the pressure and temperature limitations of these MODES. For these events, the CCS System is not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1

With one CCS train inoperable, the inoperable CCS train must be restored to OPERABLE status within the Completion Time of 72 hours before action must be taken to reduce power. The specified Completion Time is consistent with other LCOs for the loss of one train of an ESF system. The Completion Time of 72 hours is based on industry-accepted practice, the low probability of an accident occurring during the 72 hours period while one CCS train is inoperable, and engineering judgment considering the number of available systems and the time required to reasonably complete the Required Action.

B.1 and B.2

If the inoperable CCS System train cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cool down in such a time frame on one safety system train.

C.1 and C.2

With both CCS System trains inoperable, the unit is not prepared to respond to the design basis events for which CCS is required. The unit must be placed in a MODE in which the risk to the unit and to the environment is minimized. Without CCS being supplied to the Reactor Coolant Pumps (RCPs), the unit will not continue power operation and will, by procedure, be brought to MODE 3 quickly to prevent damage to the RCPs. Depending on the actual state of the CCS System, the RHR System may not have a safety-grade heat sink and the unit should be placed in MODE 4, a condition where decay heat can be removed by the steam generators. Twelve hours is a reasonable time, based on operating experience, to place the unit in MODE 4 from full power conditions without challenging safety systems or operators.

Action should be taken within one hour to place the unit in MODE 5, however, if the CCS System is not operating at all, the unit may choose to remain in MODE 4. If CCS flow is available to the RHR heat exchangers to further cool the unit, unit cool down should continue.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Verifying the correct alignment of manual, power-operated, and automatic valves in the CCS System provides assurance that the proper flow path exists for CCS operation. This surveillance requirement does not apply to valves which are locked, sealed, or otherwise secured in position or to check valves. The surveillance interval of 31 days is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.8.2

For the CCS System to provide a continuous supply of cooling water to safety-related equipment and to isolate the nonessential segments of the system following a LOCA or MSLB, automatic valves must position on the actuation signals. This surveillance ensures that each automatic valve actuates on a [Phase A Containment Isolation, Phase B Containment Isolation, and Surge Tank Level--Low] Signal[s]. The surveillance interval of 18 months is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.8.3

For the CCS System to provide a continuous supply of cooling water to safety-related equipment following a LOCA or MSLB, the CCS pumps must start on the actuation signals. This surveillance ensures that the pumps start on a [Safety Injection] signal. The surveillance interval of 18 months is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.8.4

This surveillance ensures that the C-S pump is powered from the appropriate AC Power Source when aligned for OPERABLE status. The surveillance frequency is based on engineering judgement.

BASES (continued)

REFERENCES

1. Watts Bar FSAR Section [9.2.2], Component Cooling Water System.

2. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 44, "Cooling Water.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.9 Essential Raw Cooling Water (ERCW)

BASES

BACKGROUND

The ERCW System provides both safety-related cooling and nonsafety-related cooling. ERCW is used during normal operating and normal shutdown conditions and provides cooling to both the safety-related and nonsafety-related cooling loads. The safety-related portion of the ERCW System is used during normal shutdown conditions and abnormal conditions, such as loss of offsite power and a Loss of Coolant Accident (LOCA) or a Main Steam Line Break (MSLB).

The safety-related portion of the ERCW System is covered by this LCO. The safety-related portion of the ERCW System circulates cooling water from the Ultimate Heat Sink (UHS) through the unit components required for safe shutdown of the reactor following an accident, and discharges the water to the cooling tower basins or to the holding pond. [The ERCW also provides emergency makeup to the Component Cooling Water (CCS) System and is the backup water supply to the Auxiliary Feedwater System.]

The ERCW system consists of eight ERCW pumps, four traveling water screens, four screen wash pumps, four strainers, associated piping, valves, and instrumentation.

Water for the ERCW system enters two separate sump areas of the pumping station through four traveling water screens, two for each sump. Four ERCW pumping units, all on the same plant train, take suction from one of the sumps, and four more on the opposite plant train take suction from the other sump. One set of pumps and associated equipment is designated Train A, and the other Train B. These trains are redundant and are normally maintained separate and independent of each other. Each set of four pumps discharges into a common manifold, from which two separate headers (1A and 2A for Train A, 1B and 2B for Train B), each with its own automatic backwashing strainer, supply water to the various system users.

(continued)

BASES (continued)

BACKGROUND (continued)

Minimum combined safety requirements for one 'accident' unit and one 'non-accident' unit, or two 'non-accident' units, are met by only two pumps on the same plant train. Since the A and B ERCW plant trains, which are shared between the units, each have two pumps that are assigned to the emergency diesel generators on loss of offsite power, total loss of either header, or the loss of offsite power and an entire plant emergency power train will not prevent safe shutdown of either unit under any credible plant condition. Therefore, the sharing of the ERCW system between the two units does not compromise safety.

Each ERCW train provides separate cooling to duplicate components which are considered essential to unit cooldown or for cooling requirements to support recovery following transient and accident conditions. Included among the supported components is the diesel generator. The ERCW is considered an ESF System because it is vital during recovery from an accident. To this end, the system is designed to meet Class I requirements for structural integrity and reliability required by 10 CFR 50. Although not explicitly covered by this LCO, the ERCW may be used for unit cooldown and during unit shutdown to provide cooling water to the CCS heat exchangers during MODES 5 and 6.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the ERCW ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The ERCW is designed to remove heat from components important to mitigate the consequences of a LOCA or MSLB. The redundant cooling capacity of this system and each of the safety-related systems which it serves, assuming a single failure coincident with the loss of offsite power, is consistent with the assumptions used in the safety analysis (Ref. 1). All vital power can be supplied from either onsite or offsite power systems.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

Determination of the minimum flows to each of the required cooling loads is based on a maximum UHS temperature of [85]°F. The ERCW design assures that the flow requirements are achieved by operation of two ERCW pumps on one train and proper realignment of the valves to the accident configuration (Ref. 2).

The ERCW is a support system to various systems which are assumed to function to mitigate various Design Basis Accidents (DBAs). The ERCW is a system that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, the ERCW satisfies the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The ERCW trains are independent of each other to the degree that each has separate controls and power supplies, and the operation of one does not depend on the other. In the event of a LOCA or MSLB, one train of ERCW is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of ERCW must be OPERABLE. At least one train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

APPLICABILITY

The ERCW ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal operations and during and after a LOCA or MSLB accident. In MODES 1, 2, 3, and 4, a LOCA or MSLB could demand the systems supported by ERCW. The possibility of these events in MODES 5 and 6 is low due to the pressure and temperature limitations of these MODES. For these events, the ERCW is not required to be OPERABLE in MODES 5 and 6. Components that are required OPERABLE in MODES 5 and 6 and rely on ERCW flow are covered by the definition of OPERABILITY to that system.

(continued)

BASES (continued)

ACTIONS

A.1

With one ERCW train inoperable, the inoperable ERCW train must be restored to OPERABLE status within 72 hours before action must be taken to reduce power. The specified Completion Time is consistent with other LCOs for the loss of one train of an ESF System. The Completion Time of 72 hours is based on industry-accepted practice, the low probability of an accident occurring during the 72 hours while one ERCW train is inoperable, and engineering judgment considering the number of available systems and the time required to reasonably complete the Required Action.

B.1 and B.2

If the inoperable ERCW train cannot be restored in the required Completion Time, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cooldown in such a time frame on one safety system train.

C.1 and C.2

With both ERCW System trains inoperable, the unit is not prepared to respond to the design basis events for which the ERCW System is required. The unit must be placed in a MODE in which the risk to the unit and to the environment is minimized. Depending on the actual state of the ERCW System, the RHR System may not have a safety-grade heat sink and the unit should be placed in MODE 4, a condition where decay heat can be removed by the steam generators. Twelve hours is a reasonable time, based on industry operating experience, to place the unit in MODE 4 from full power conditions without challenging safety systems or operators.

Action should be initiated within one hour to place the unit in MODE 5. However, in this extraordinary situation, the unit may choose to remain in MODE 4 until a method to further cool the unit is available. If ERCW flow is available to the CCS heat exchanger to further cool the unit via the RHR heat exchangers, unit cooldown should continue.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

Verifying the correct alignment of manual, power-operated, and automatic valves in the ERCW System provides assurance that the proper flow path exists for ERCW System operation. This surveillance requirement does not apply to valves which are locked, sealed, or otherwise secured in position or to check valves. The surveillance interval of 31 days is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.9.2

For the ERCW System to provide a supply of cooling water to safety-related equipment [and to isolate the nonsafety-related portion of the ERCW System following a LOCA or MSLB, automatic valves must position on the actuation signals.] This surveillance ensures that each automatic valve actuates on an [ESFAS] signal. The surveillance interval of 18 months is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

SR 3.7.9.3

For the ERCW System to provide a supply of cooling water to safety-related equipment following a LOCA or MSLB, the ERCW pumps must start on the actuation signals. This surveillance ensures that the pumps start on a [Safety Injection] signal. The surveillance interval of 18 months is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 44, "Cooling Water."

2. Watts Bar FSAR Section [9.2.1] ERCW System.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.10 Ultimate Heat Sink

BASES

BACKGROUND

The Ultimate Heat Sink is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads. The maximum temperature of the Ultimate Heat Sink ensure adequate heat load removal capacity for a minimum of 30 days after a normal reactor shutdown or a shutdown following an accident, including a Loss of Coolant Accident (LOCA). The water source which makes up the Ultimate Heat Sink is the Tennessee River. The Essential Raw Cooling Water (ERCW) System is designed to operate at a maximum temperature of [85]°F. The Ultimate Heat Sink complete system description and how the Ultimate Heat Sink complies with the requirements of Regulatory Guide 1.27, Rev. [1], (Ref. 1) are described in the Final Safety Analysis Report (FSAR) Section [9.2.5] (Ref. 2).

APPLICABLE SAFETY ANALYSES

The parameters of the Ultimate Heat Sink are not modeled in any safety analysis. However, the assumption is made that there is an Ultimate Heat Sink which provides cooling water for accident mitigation. Thus, the Ultimate Heat Sink serves as a support system to various systems which are assumed to function to mitigate various Design Bases Accidents (DBA). The Ultimate Heat Sink is a system that is part of the primary success path and which functions or actuates to 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. As such, the Ultimate Heat Sink satisfies Criterion 3 of the NRC Interim Policy Statement (Ref.3).

(continued)

BASES (continued)

LCOs

Without an OPERABLE Ultimate Heat Sink, the heat generated from fission product decay following shutdown may not be removed and could result in the eventual melting of the fuel elements and release of fission product activity to the reactor coolant. Should this occur in conjunction with a LOCA, fission product activity could be released to the containment atmosphere.

APPLICABILITY

The Ultimate Heat Sink ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal operations and during and after a DBA. The Ultimate Heat Sink must be OPERABLE in MODES 1, 2, 3, and 4, when a DBA could demand substantial heat removal capacity. In MODES 5 and 6, the heat load is significantly reduced and as such the Ultimate Heat Sink is not specifically required to be OPERABLE. Decay heat removal in MODES 5 and 6 is addressed by LCO 3.4.7 RCS Loops - MODE 5, Loops Filled, LCO 3.4.8 RCS Loops - MODE 5, Not Filled, LCO 3.9.5 Residual Heat Removal and Coolant Circulation - High Water Level, and LCO 3.9.6 Residual Heat Removal and Coolant Circulation - Low Water Level.

ACTIONS

A.1 and A.2

If the Surveillance Requirements are not met or other conditions cause the Ultimate Heat Sink to be declared inoperable, the unit must be placed in a MODE where the requirement does not apply. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the Completion Time of 36 hours to reach MODE 5 is reasonable considering that a unit can easily cool down in such a time frame on one safety system train. In MODE 5 the Ultimate Heat Sink is no longer required to be OPERABLE due to the pressure and temperature limitations of this MODE except to support equipment required to be OPERABLE such as the RHR System.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

The temperature specified ensures that the Essential Raw Cooling Water (ERCW) System can cool the Component Cooling Water System and other safety related heat loads to at least its maximum design temperature with the maximum accident or normal design heat loads. The frequency of [24] hours is based on engineering judgment. This frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. Regulatory Guide 1.27, Rev.[1], Ultimate Heat Sink for Nuclear Power Plants, .
2. Watts Bar, FSAR, Section [9.2.5], Ultimate Heat Sink.
3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.11 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool is required to meet the Iodine decontamination factor assumptions following a fuel handling accident. The specified water level provides shielding to minimize the general area dose with the fuel storage racks filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. Should normal cooling be lost, the water provides sufficient heat capacity that boiling will not occur for approximately [6] hours.

A general description of the fuel storage pool design is provided in the Final Safety Analysis Report (FSAR) Section [9.1.2] (Ref. 1). This design is in accordance with Regulatory Guide (RG) 1.13 (Ref. 2). The assumptions of the fuel handling accident are found in FSAR Section 15.5.6 (Ref. 3). These assumptions are in agreement with RG 1.25 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The assumption of RG 1.25, preserved by this specification, is that there is $\geq$ 23 feet of water between the top of the damaged fuel bundle and the fuel pool surface. With 23 feet, the assumption of RG 1.25 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. However, in the case of a single bundle dropped and lying horizontally on top of the spent fuel racks there may be less than 23 feet between the top of the fuel bundle and the surface by the width of the bundle. To offset this small non-conservatism, the analysis assumes that [all] fuel rods fail. The resultant 2 hour thyroid dose to a person at the exclusion area boundary is well within the 10 CFR 100 exposure guidelines (Ref. 5).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	The minimum fuel storage pool water level is a process variable that is an initial condition of a Design Basis Accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission barrier. As such, it satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 6).
LCOs	The specified water level is the minimum level required to meet the assumptions of the fuel handling accident.
APPLICABILITY	This LCO is only applicable when irradiated fuel is being moved in the fuel storage pool since the potential for a significant release of fission products can only exist with irradiated fuel in the pool. LCOs 3.0.3 and 3.0.4 are not applicable as events in the fuel storage pool are not affected by either MODE level or unit operations.
ACTIONS	<u>A.1 and A.2</u> With the fuel storage pool, water level less than 23 feet above the top of irradiated fuel assemblies seated in storage racks, the movement of fuel is immediately stopped. This suspension of either fuel movement or crane operations shall not preclude the movement of these loads to a safety, conservative position. This effectively precludes the possibility of a spent fuel handling accident. Crane operations are covered separately by the NUREG 0612 Heavy Load Program. Action should be initiated immediately to restore fuel storage pool water level and action continued as required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

The water level in the fuel storage pool must be checked periodically. The surveillance frequency of 7 days is based upon engineering judgment and extensive unit experience. [When refueling operations are taking place, the level in the pool is at equilibrium with the level in the refueling cavity, and the refueling cavity level is verified daily.]

REFERENCES

1. Watts Bar FSAR, Section [9.1.2], Spent Fuel Storage.
2. Regulatory Guide 1.13, Rev. [1], Spent Fuel Storage Facility Design Basis, [December, 1975].
3. Watts Bar FSAR, Section [15.5.6], Fuel Handling Accident.
4. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, [March 23, 1972].
5. Title 10 Code of Federal Regulations, Part 100, Reactor Site Criteria.
6. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," United States Nuclear Regulatory Commission, February 6, 1987.

B 3.7 PLANT SYSTEMS

B 3.7.12 Atmospheric Relief Valves (ARVs)

BASES

BACKGROUND

One ARV and its associated controls is provided on the main steamline for each steam generator on the MSSV header. The ARVs serve to provide the capability for steam generator pressure control and for a controlled plant cooldown when steam dump to the main condenser is not available. A manual isolation valve is provided between the ARVs and the main steamline to facilitate maintenance of these valves. Two safety grade solenoid valves are provided in the control loops for these ARVs. The controls for these solenoid valves, coupled with the failed closed position of the ARVs, provide assurance that the ARVs will only be opened when needed for plant cooldown or mitigation of plant steamline pressure transients.

Operator action is required for the correct operation of the ARVs. In the event the ARVs are used for cooldown to a hot shutdown (RHR cut-in condition), the operator will manually adjust the ARVs pressure setpoint to maintain the rate of the cooldown within the desired range of approximately 50°F per hour. The manual control of the ARVs can be accomplished locally at the valves or from the main or auxiliary control room. The operator may override the operation of the ARVs with safety grade open/close handswitches located in the MCR.

APPLICABLE SAFETY ANALYSES

The atmospheric relief valves provide the means for plant cooldown by steam discharge to the atmosphere if the condenser steam dump is not available. The valves will also provide a means of steam generator pressure control if the condenser steam dump is not available, and will thus preclude unnecessary lifting of the MSSVs.

Cooldown may be required following expected transients or following an accident such as a main steamline break, small break LOCA, or steam generator tube rupture.

LCOs

The ARVs must be OPERABLE whenever significant mass and energy are present in the RCS and steam generators to provide a means for plant cooldown.

(continued)

BASES (continued)

APPLICABILITY

The ARVs provide the means for plant cooldown for MODES 1, 2, and 3 should the preferred condenser not be available.

The ARVs need not be OPERABLE in MODES 4, 5, and 6 because the RHR system will normally be used for further plant cooldown.

ACTIONS

A.1

With one ARV inoperable, time is allowed for repair. The Completion Time of 7 days is based on engineering judgement and industry operating experience.

B.1

With more than one ARV inoperable, at least three ARVs are required to be restored to OPERABLE status within 24 hours. since only two ARVs are required to meet the accident analyses. The Completion Time is considered reasonable.

C.1 and C.2

If the Required Actions cannot be met within the required Completion Times, a controlled shutdown is required. The Completion Time of 6 hours specified for MODE 3 and 12 hours for MODE 4 is based on engineering judgement and industry operating experience without challenging either safety systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Demonstrating the ARVs can be cycled ensures their availability if required for a plant cooldown without the main condenser. The frequency of 18 months is based on engineering judgement.

REFERENCES

1. Watts Bar FSAR, Chapter 10.2.

B 3.7 PLANT SYSTEMS

B 3.7.13 Control Room Emergency Ventilation System

BASES

BACKGROUND

The CREVS provides the control room with a conditioned atmosphere following various Design Basis Accidents (DBAs) such as Loss of Coolant Accident (LOCA), fuel handling accident, rod ejection, main steamline break and steam generator tube rupture.

The CREVS consists of two separate and redundant 100% capacity trains which recirculate the control room air. The system design complies with 10 CFR 50, Appendix A, General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Bases", GDC 19, "Control Room", (Ref. 1), and Regulatory Guide 1.78 (Ref. 2) requirements for control room habitability systems and is described in FSAR Section [6.4] (Ref. 3).

The Control Room Emergency Ventilation System includes two separate and redundant trains of filtered outside air. Each train consists of a high efficiency particulate air (HEPA) filter bank, an activated charcoal adsorber section for removal of gaseous activity, an air cleanup unit fan, emergency pressurization fan, and associated air handling unit. Ductwork, valves and/or dampers and instrumentation also form part of the system. The system initiates filtered ventilation of the control room following receipt of a [Control Room Isolation] signal. A reduced flow of outside air is filtered and added to the air being recirculated from the control room. The addition of outside air by redundant emergency pressurizing fans pressurizes the control room to a minimum of [1/8] inches water gage. Pressurization of the control room minimizes infiltration of unfiltered air from surrounding areas of the building.

A Control Room Isolation Signal closes the unfiltered outside air supply and aligns the system to the isolation mode. The control room air is recirculated through redundant trains of heating/air-conditioning, and a portion thru HEPA filters, and charcoal filters. Redundant paths of filtered outside air are aligned to add air to the control room, pressurizing the control room.

(continued)

BASES (continued)

BACKGROUND (continued)

The system initiates filtered ventilation of the control room following receipt of a High Air Intake Radiation Level, High Smoke Density, or a Safety Injection Signal.

OPERABILITY of the Control Room Emergency Air Cleanup System assures that the control room will be maintained in a safe and habitable condition under accident conditions by providing protection against radiation in accordance with GDC 4 and 19 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CREVS design basis is established by the consequences of the limiting DBA which is a LOCA in MODES 1, 2, 3 and 4 [and which is a fuel handling accident in MODES 5 and 6]. The LOCA analysis (Ref. 4) assumes that only one train of the CREVS is functional due to a single failure which disables the other train. The accident analysis accounts for the reduction in radioactive material provided by the remaining train. The amount of fission products released from containment is determined for a LOCA. The calculated control room dose meets the guideline values of GDC 19 (Ref. 1).

The CREVS is part of the primary success path which functions or actuates to 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. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5).

LCOs

The CREVS trains have separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of the CREVS is required to provide the minimum fission product removal assumed in the safety analysis. Two trains of the CREVS must be OPERABLE to ensure that these minimum requirements are met. This will ensure that at least one train will operate, assuming that the other train is disabled by a single failure.

(continued)

BASES (continued)

APPLICABILITY

The CREVS is required to be OPERABLE in MODES 1, 2, 3, and 4 to provide isolation from and removal of fission products to assure that the control room remains habitable following a LOCA.

The CREVS is required to be OPERABLE in MODES 5 and 6 to assure that the control room remains habitable following failure of a waste gas decay tank containing radioactive material.

The CREVS is required to be OPERABLE during movement of irradiated fuel to assure that the control room remains habitable following a fuel handling accident.

ACTIONS

A.1

When one train of the CREVS is determined to be inoperable, action is required to restore the train to OPERABLE status. A Completion Time of 7 days is permitted to restore the system to OPERABLE status before action must be taken to reduce power. This Completion Time of 7 days is based on engineering judgement, considering that the remaining train can provide all the required capabilities and provides adequate time to make most repairs.

B.1 and B.2

If the inoperable train(s) of the CREVS can not be restored in the required Completion Time or if two CREVS trains are inoperable during MODES 1-4, the unit must be placed in a MODE where a DBA is not credible. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is based on normal cooldown rates.

(continued)

BASES (continued)

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

If the Required Actions and associated Completion Times of Condition A are not met in MODE 5 and 6 or during movement of irradiated fuel, the OPERABLE train must be placed in the [isolation] mode within 7 days or CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel must be suspended Immediately.

D.2.1, D.2.2, and D.2.3

If two trains of the Control Room Emergency Ventilation System are inoperable in MODES 5 and 6 or during movement of irradiated fuel, then all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel must be suspended Immediately. For these Conditions neither train is considered to be operational. Depending on the reasons for the trains' inoperability the system may not be able to maintain habitable conditions in case of an accident or to maintain the control room temperature within equipment environmental qualification limits. In either case actions must be taken to preclude the occurrence of events that could require the use of the CREVS.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This test provides periodic confirmation of the OPERABILITY of each train of the Control Room Air Cleanup System. The frequency of 31 days is based on engineering judgement taking into account the normally mild environment and operating conditions and the low likelihood of failure on demand. Additionally, this frequency has been shown to be acceptable through industry operating experience.

SR 3.7.13.2

The required CREVS filter testing will be performed in accordance with the Ventilation Filter Testing Program.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

The automatic startup test verifies that both trains of equipment start on receipt of each actual or simulated Control Room Isolation signal. The frequency of 18 months is considered to be adequate considering the high reliability of electronic actuation systems.

SR 3.7.13.4

The control room positive pressure is periodically tested to verify proper function of the CREVS and to verify the integrity of the control room enclosure. The frequency of 18 months is consistent with the guidance provided in NUREG 0800, Section 6.4 (Ref. 6).

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants;
   Criterion 4, "Environmental and Dynamic Effects Design Bases", and
   Criterion 19, "Control Room."

2. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

3. Watts Bar FSAR, Section [6.5], "Fission Product Removal and Control Systems".

4. Watts Bar FSAR, Chapter [15], "Accident Analysis".

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements For Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

6. NUREG-0800, "Standard Review Plan", Section 6.4, [Rev. 2, July 1981], "Control Room Habitability System".

B 3.7 PLANT SYSTEMS

B 3.7.14 Control Room Emergency Air Temperature Control (HVAC) System

BASES

BACKGROUND

The Control Room Emergency Air Temperature Control System provides the control room with a conditioned atmosphere following various Design Basis Accidents (DBAs) such as Loss of Coolant Accident (LOCA), fuel handling accident, rod ejection, main steamline break and steam generator tube rupture. This system ensures that the instrumentation and equipment located in the control room will be maintained within their design temperatures and that the control room will remain habitable during both normal and isolated modes of operation.

The Control Room Emergency Air Temperature Control System consists of two trains, with each train designed to provide 100% cooling capacity for the Main Control Room Habitability Zone (MCRHZ). Each train consists of an air-handling unit (AHU), a water chiller, a chilled-water pump, and associated piping, ductwork, dampers, instrumentation and controls.

APPLICABLE SAFETY ANALYSES

The Control Room Emergency Air Temperature Control System design basis is established by the consequences of the limiting DBA which is a LOCA in MODES 1, 2, 3 and 4 [and which is a fuel handling accident in MODES 5 and 6]. The LOCA analysis (Ref. 1) assumes that only one train is functional due to a single failure which disables the other train.

The Control Room Emergency Air Temperature Control System is part of the primary success path which functions or actuates to 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. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 2).

(continued)

BASES (continued)

LCOs

The Control Room Emergency Air Temperature Control trains have separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of the Control Room Emergency Air Temperature Control is required to control the temperature to within environmentally qualified limits of safety-related equipment operation. Two trains of the Control Room Emergency Air Temperature Control must be OPERABLE to ensure that these minimum requirements are met. This will ensure that at least one train will operate, assuming that the other train is disabled by a single failure.

APPLICABILITY

The Control Room Emergency Air Temperature Control System is required to be OPERABLE in MODES 1, 2, 3, and 4 to assure that the control room remains habitable following a LOCA.

The Control Room Emergency Air Temperature Control System is required to be OPERABLE in MODES 5 and 6 to assure that the control room remains habitable following failure of a waste gas decay tank containing radioactive material.

The Control Room Emergency Air Temperature Control System is required to be OPERABLE during movement of loads over irradiated fuel or during movement of irradiated fuel to assure that the control room remains habitable following a fuel handling accident.

ACTIONS

A.1

When one train of the Control Room Emergency Air Temperature Control is determined to be inoperable, action is required to restore the train to OPERABLE status. A Completion Time of 30 days is permitted to restore the system to OPERABLE status before action must be taken to reduce power. This Completion Time of 30 days is based on engineering judgement, considering that the remaining train can provide all the required capabilities and provides adequate time to make most repairs.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If the inoperable train can not be restored in the required Completion Time in MODES 1-4, the unit must be placed in a MODE where a DBA is not credible. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The Completion Time of 6 hours is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging safety systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is based on normal cooldown rates.

C.1, C.2.1, C.2.2, and C.2.3

If the Required Actions and associated Completion Times of Condition A are not met in MODE 5 and 6 or during movement of irradiated fuel, the OPERABLE train must be placed in operation within 7 days or CORE ALTERATIONS, positive reactivity additions, and movement of irradiated fuel must be suspended Immediately.

D.2.1, D.2.2, and D.2.3

If two trains of the Control Room Emergency Air Temperature Control System are inoperable in MODES 5 and 6 or during movement of irradiated fuel, then all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel must be suspended Immediately. For these Conditions neither train is considered to be operational. Depending on the reasons for the trains' inoperability the system may not be able to maintain habitable conditions in case of an accident or to maintain the control room temperature within equipment environmental qualification limits. In either case actions must be taken to preclude the occurrence of events that could require the use of the system.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This test provides periodic confirmation of the OPERABILITY of each train of the Control Room Emergency Air Temperature Control System. The frequency of 31 days is based on engineering judgement taking into account the normally mild environment and operating conditions and the low likelihood of failure on demand. Additionally, this frequency has been shown to be acceptable through industry operating experience.

REFERENCES

1. Watts Bar FSAR, Chapter [15], "Accident Analysis".
2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements For Nuclear Power Reactors", United States Nuclear Regulatory commission, February 6, 1987.
3. NUREG-0800, "Standard Review Plan", Section 6.4, [Rev. 2, July 1981], "Control Room Habitability System".

B 3.7 PLANT SYSTEMS

B 3.7.15 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

The ABGTS has primarily two functions. It ensures that radioactive materials coming from leaking ECCS equipment in the pump rooms following a LOCA, or an irradiated fuel assembly that is damaged during a fuel handling accident, are filtered prior to discharge to the environment. This system reduces the potential release of radioactive material to within values specified in 10 CFR 50 (Ref. 1).

The ABGTS consists of two separate and redundant trains. Each train includes [a demister, heater, prefilter, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity, principally iodines, another HEPA filter, and a fan.] Ductwork, valves and/or dampers and instrumentation also form part of the system. [The heater functions to reduce the relative humidity of the air stream.] A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. No credit is taken for the downstream HEPA in the safety analysis. The system initiates filtered ventilation of the Auxiliary Building Secondary Containment enclosure (ABSCE) following receipt of a Phase A isolation or High radiation signal. The system design complies with the 10 CFR 50, Appendix A, General Design Criteria (GDC) 41, 42, and 43 requirements for containment cleanup systems and 61 and 64 for Fuel Building Air Cleanup Systems (Ref.2) and is described in FSAR Section [6.5] (Ref.3).

[The demister is included for moisture (free water) removal from the gas stream.] The heaters are used to heat the gas stream which lowers the relative humidity. Continuous operation of each train for at least 10 hours per month with the heaters on reduces moisture buildup on (drys) the HEPA filters and adsorbers. Both the demister and heater are important to the effectiveness of the charcoal adsorbers.]

The ABGTS reduces the radioactive content of the ABSCE exhaust to the atmosphere following a DBA. Loss of the ABGTS could cause SITE BOUNDARY doses, in the event of a DBA, to exceed the values given in 10 CFR 100 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting DBA which is a fuel handling accident or a LOCA. The accident analysis, (Ref. 4) of the fuel handling assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assures that radioactive materials leaked from the ECCS are filtered and adsorbed by the ABGTS. The analysis assumes that only one train of the ABGTS is functional due to a single failure which disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from ECCS equipment in the ECCS pump room is determined for a LOCA and from the fuel handling area for a fuel handling accident.

The modeled ABGTS actuation in the safety analysis is based upon a worse case response time associated with a LOCA. The results indicate the ABGTS can reach and maintain a negative 1/4-inch water gauge pressure in the ABSCE within [4] minutes of the occurrence of a LOCA.

The ABGTS is part of the primary success path which functions or actuates to 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. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5).

LCOs

The ABGTS trains have separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one train of the ABGTS is required to provide the minimum fission product removal assumed in the safety analysis. Two trains of the ABGTS must be OPERABLE to ensure that these minimum requirements are met. This will ensure that at least one train will operate, assuming that one train is disabled by a single failure.

APPLICABILITY

LCO 3.7.15 requires the ABGTS to be OPERABLE in MODES 1,2, 3, and 4 because of the potential for a fission product release following a DBA. It also has to be OPERABLE anytime there is movement of irradiated fuel in the fuel handling area in case of a fuel handling accident.

(continued)

BASES (continued)

ACTIONS

A.1

When one train of the ABGTS is determined to be inoperable, action is required to restore the system to OPERABLE status. A Completion Time of 7 days is permitted to restore the system to OPERABLE status before action must be taken to reduce power. This Completion Time is based on engineering judgement, considering that the remaining train can provide all of the required air cleanup capability.

B.1 and B.2

If the inoperable train of the ABGTS cannot be restored in the required Completion Time in MODES 1-4, the unit must be placed in a MODE in which a DBA is not credible. This is done by placing the unit in MODE 3 in 6 hours and in MODE 5 in the next 30 hours. The 6 hours allowed is a reasonable time, based on industry operating experience, to reach MODE 3 from full power without challenging unit systems or operators. Similarly, the 30 hours allowed for reaching MODE 5 is reasonable considering that a unit can cooldown in such a time frame on one safety system train.

C.1

If the Required Actions for Condition A or B are not met in the required Completion Times, or if one train of ABGTS is inoperable during movement of irradiated fuel in the fuel handling area, the OPERABLE train must be placed in operation immediately, or movement of fuel in the fuel handling area suspended.

D.1

With two trains of ABGTS inoperable, movement of irradiated fuel in the fuel handling area must be immediately suspended. The note says that LCO 3.0.3 is not applicable because the actions in 3.0.3 are not appropriate for this specific condition.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Continuous operation of the system for 10 hours with the heaters on assures that the filters and charcoal adsorbers are dry for maximum effectiveness. The frequency of 31 days is based on engineering judgement taking into account the importance of the system, the normally mild environment and operating conditions and the low likelihood of failure on demand. Additionally, this frequency has been shown to be acceptable through industry operating experience.

SR 3.7.15.2

Filter testing is performed in accordance with the Ventilation Filter Testing Program in Specification 5.9.13.

SR 3.7.15.3

The automatic startup test verifies that both trains of equipment start on receipt of an actual or simulated actuation test signal. The frequency of 18 months is considered to be adequate considering the reliability of electronic actuation systems. Additionally, this frequency has been shown to be acceptable through industry operating experience.

SR 3.7.15.4

The areas serviced by the ABGTS are periodically tested to verify proper performance of the ABGTS and to verify that the auxiliary building secondary containment enclosure (ABSCE) is maintained at a negative pressure to prevent out-leakage to the atmosphere. The frequency of 18 months is consistent with the Regulatory Guide 1.52 (Ref. 8) guidance.

SR 3.7.15.5

The ABGTS/ABSCE dampers are tested to verify OPERABILITY. The ABGTS/ABSCE dampers are in a different position during normal operation and must reposition for accident operation to draw air through the filters. The surveillance frequency of 18 months is considered to be acceptable based on the damper reliability and design. Additionally, this frequency has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100, 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance".

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants;
   Criterion 41, "Containment atmosphere cleanup",
   Criterion 42, "Inspection of containment atmosphere cleanup systems",
   Criterion 43, "Testing of containment atmosphere cleanup systems".
   Criterion 61, "Fuel Storage and Handling and Radioactivity Control", and
   Criterion 64, "Monitoring Radioactivity Releases".

3. Watts Bar FSAR, Section [6.5], "Fission Product Removal and Control Systems".

4. Watts Bar FSAR, Chapter [15], "Accident Analysis".

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The AC power system consists of the offsite power sources, (preferred power), and the onsite standby power sources. As required by 10CFR50, Appendix A, General Design Criteria 17, "Electric power systems", (Ref. 1), the design of the AC power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The safety-related station loads are divided into two trains, with each train fed by an ESF bus. Each ESF bus has two separate and independent offsite sources of power. Either of the two trains provide for the minimum safety functions necessary to shutdown the unit and maintain it in a safe shutdown condition. An electrical power distribution system diagram is provided in Figure B 3.8.1-1.

AC power from the [161] kV transformer yard to the onsite electrical distribution system is supplied by two independent circuits. The two independent circuits, supply the 6900 VAC ESF buses

The normal feeds during unit operation are:

- From Unit Station Service transformer (USST) 1A through breaker 1114 to Unit Board 1B through breaker 1712 to breaker 1718 on 6.9 kV Shutdown Board 1A-A.

- From USST 1B through breaker 1122 to Unit Board 1C through breaker 1722 to breaker 1726 on 6.9 kV Shutdown Board 1B-B.

- From USST 2A through breaker 1214 to Unit Board 28 through breaker 1814 to breaker 1818 on 6.9 kV Shutdown Board 2A-A.

- From USST 2B through breaker 1222 to Unit board 2C through breaker 1822 to breaker 1826 on 6.9 kV Shutdown Board 2B-B.

The unit boards may also be manually or automatically transferred to offsite power.

(continued)

BASES (continued)

BACKGROUND (continued)

The first alternates (offsite power) are:

- From common station service transformer (CSST) C breaker 1712 to breaker 1716 to 6.9 kV Shutdown Board 1A-A.
- From CSST D breaker 2814 to breaker 1728 to 6.9 kV Shutdown Board 1B-B.
- From CSST transformer C breaker 1712 to breaker 1816 to 6.9 kV Shutdown Board 2A-A.
- From CSST D breaker 2814 to breaker 1828 to 6.9 kV Shutdown Board 2B-B.

A single offsite circuit is capable of providing the ESF loads. Both of these circuits are required to meet the Limiting Condition for Operation.

The on-site standby power source for each 6900 VAC ESF bus is a diesel generator. WBN uses 4 diesel generator sets for Unit 1 operation. These same diesel generators will be shared for Unit 2 operation. WBN also utilizes a C-S diesel generator that can be aligned to any 6.9 kV ESF bus. The diesel generator starts automatically on safety injection signal, or a loss of offsite power, e.g. bus loss of voltage.

Ratings for train 1A, 1B, 2A, and 2B diesel generators, satisfy the requirements of Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", (Ref. 2). The continuous service rating of each of the diesel generators is [4400] kW.Each diesel generator set also has an additional rating of 4840 kW for 2 hours out of 24. The ESF loads which are powered from the [6.9] kV emergency buses are listed in Reference 3.

The plant [6.9] kV buses provide AC electrical power for the safety related equipment. A decrease or loss of offsite power on the bus is detected by degraded and loss of voltage for initiation of the load shedding and sequencing logic which results in offsite power circuit breaker trip, diesel generator start and load sequencing. The under voltage relays sense loss of voltage to a level at which electrical equipment would not function. The degraded voltage relay(s) trip(s) when the voltage condition is such that continuous operation of the equipment is undesirable.

(continued)

"This Figure For Illustration Only Do Not Use For Operation"

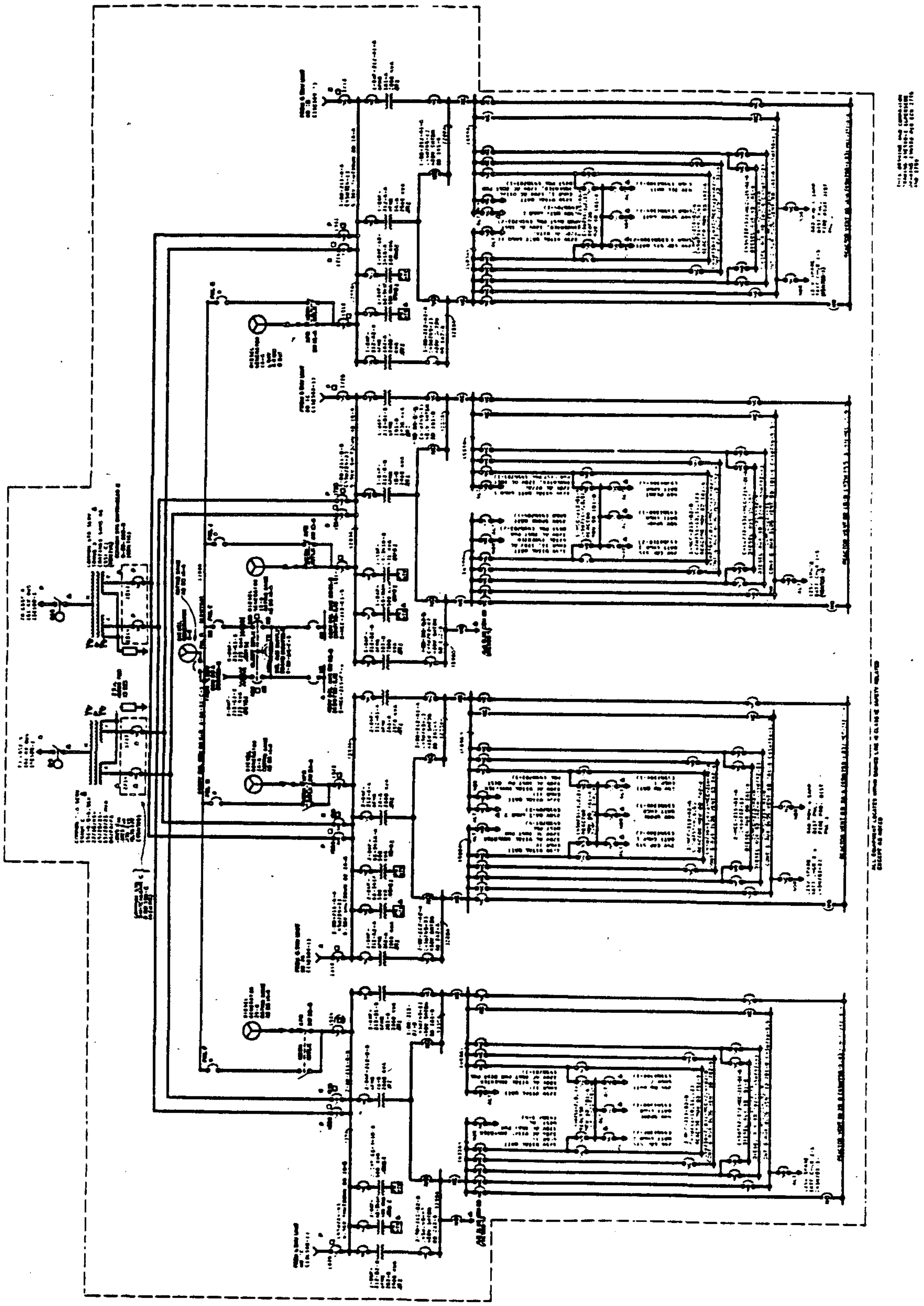

Figure B 3.8.1-1 (Page 1 of 1)
Electrical Power System

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of design basis transient and accident analyses in FSAR Chapters [6] and [15] assume all ESF systems are OPERABLE. The AC power system is 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, assuming a loss of offsite power and a single failure in the onsite emergency power system.

The OPERABILITY of the power sources are consistent with the initial assumptions of the accident analyses and are based upon maintaining at least train A or B of the onsite AC and DC power sources and associated distribution systems OPERABLE during accident conditions, assuming a loss of offsite power, and a single failure.

The AC Power Source System is part of the primary success path which functions or actuates to mitigate a MODE 1, 2, 3, or 4 Design Basis Accident (DBA) or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

LCOs

Both of the physically independent circuits between the offsite transmission network and the onsite safety-related distribution system, and the standby diesel generators ensure availability of the required power to shutdown the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence or a postulated (DBA).

APPLICABILITY

The AC power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and,

(continued)

BASES (continued)

APPLICABILITY (continued)

2. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, AC Sources - Shutdown.

ACTIONS

The Required Actions specified for the levels of degradation of the power sources provide restrictions upon continued operation commensurate with the level of degradation. The C-S diesel generator may be substituted for any of the required diesel generators.

A.1 and A.2

With one of the required offsite circuits inoperable, sufficient offsite power is available from the other required offsite circuit to ensure that the unit can be shutdown and maintained in a safe shutdown condition following a design basis transient or accident. Even failure of the remaining required offsite circuit will not jeopardize a safe shutdown of the unit because of the redundant standby diesel generators. Operation could therefore safely continue if the availability of the remaining sources is verified. However, since the system reliability is degraded below the LCO requirements, a time limit on continued operation is imposed.

To ensure a highly reliable power source, it is necessary to verify the availability of the remaining required offsite circuit on a more frequent basis if one offsite circuit is inoperable. The availability of the remaining required offsite circuit must be verified within 1 hour and once per 8 hours thereafter until the inoperable offsite circuit is restored to OPERABLE status.

Per Regulatory Guide 1.93, "Availability of Electric Power Sources", (Ref. 5), operation may continue for a period that should not exceed 72 hours. If the conditions for continued power operation are met but the source is not restored within 72 hours, a controlled shutdown must be performed per Required Actions L.1 and L.2.

(continued)

BASES (continued)

ACTIONS (continued)

B.1, B.2.1, B.2.2, B.2.3.1, and B.2.3.2

In condition B only one onsite train is being powered from offsite power and one or more features powered from the train that has an offsite power available are inoperable. Failure of the remaining required offsite circuit will not jeopardize a safe shutdown of the unit because of the redundant standby diesel generators.

C.1, C.2, C.2.1, C.2.2, and C.3

With one Diesel Generator inoperable, sufficient AC power sources remain available to ensure safe shutdown of the unit in the event of a transient or accident without a single failure. Operation could therefore safely continue for a short period of time if the availability of the remaining sources is verified. Per Regulatory Guide 1.93 (Ref.5), operation may continue for a period that should not exceed 72 hours. Completion Times are consistent with those of Condition A.

In addition to verifying the availability of the offsite circuits, Required Actions C.2.1 and C.2.2 ensure the availability of the remaining diesel generators by determining the absence of a common cause failure that could affect the other diesels or by performing SR 3.8.1.2 within 24 hours. If the diesel generator is not restored to OPERABLE status within 72 hours or if Required Actions cannot be completed within their required Completion Times, a controlled shutdown must be performed per Required Actions L.1 and L.2.

D.1, D.2.1, D.2.2, D.3.1, D.3.2.1, and D.3.2.2

Condition D (loss of one diesel generator and a required feature powered from the other OPERABLE diesel generator) is a more degraded state than Condition C since loss of function may have occurred with the inoperable feature. Consequently, the Required Actions and Completion Times are consistent with Condition C except that they inoperable diesel generator or feature must be restored to OPERABLE status within 16 hours. A feature includes systems, subsystems, trains, devices and components required to perform a function.

(continued)

BASES (continued)

ACTIONS (continued)

E.1, E.2.1, E.2.2, E.3.1, and E.3.2

In Condition E, individual redundancy is lost in both the offsite power system and the onsite AC power system. However, since power system redundancy is provided by two diverse sources of power, the reliability of the power systems in this Condition may appear higher than Condition I (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. In addition to the same Required Actions of Condition C, either the offsite circuit or diesel generator must be restored to OPERABLE status. Per Reference 5, with the available offsite and diesel generator power sources each one less than required by the LCO, operation may continue for 12 hours.

If either an offsite or a diesel generator power source is restored within 12 hours, operation may continue for a total time that should not exceed 72 hours (consistent with the loss of one AC source in Condition A or C). If neither an offsite source nor a diesel generator is restored within the first 12 hours of continued power operation, a controlled shutdown must be performed per Required Actions L.1 and L.2.

F.1

New condition proposed by NRC. Bases will be provided after further discussions with NRC as to the appropriateness of the condition.

G.1, G.2.1, G.2.2, G.3.1, G.3.2, G.3.3.1, and G.3.3.2

Condition G (loss of one offsite circuit, one diesel generator and a required feature powered from the other OPERABLE AC sources) is a more degraded state than Condition E since a loss of function may have occurred with the inoperable feature. the Required Actions are consistent with the Condition E. However, the associated Completion Times are reduced. A feature includes systems, subsystems, trains, devices, and components required to perform a function.

(continued)

BASES (continued)

ACTIONS (continued)

H.1 and H.2

With two Diesel Generators inoperable, sufficient standby AC power sources are not available to feed the minimum required ESF functions with an assumed loss of offsite power.

Since the offsite power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability which could result in a total loss of AC power). However, since any inadvertent generator trip could also result in a total loss of offsite AC power, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Per Reference 5, with two diesel generators inoperable, operation may continue for a period that should not exceed 2 hours. If both diesel generators are restored within 2 hours, unrestricted operation may continue. If only one diesel generator is restored within these 2 hours, operation may continue for a total time that should not exceed 72 hours (consistent with the loss of one AC source in Condition C). If no diesel generator is restored within the first 2 hours of continued operation, a controlled shutdown must be performed per Required Actions L.1 and L.2.

I.1 and I.2

With both of the required offsite circuits inoperable, sufficient standby AC power sources are available to ensure safe shutdown of the unit in the event of a design basis transient or accident. However, since the AC power system is degraded below the LCO requirement, a time limit on continued operation is imposed.

Per Reference 5, with the available offsite AC power sources two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation may continue for a total time that should not exceed 72 hours (consistent with the loss of one offsite source in Condition A). If no offsite source is restored within the first 24 hour period of continued operation, a controlled shutdown must be performed per Required Actions L.1 and L.2.

(continued)

BASES (continued)

ACTIONS (continued)

J.1, J.2.1, and J.2.2

New condition proposed by NRC. Bases will be provided after further discussions with NRC as to the appropriateness of this condition.

K.1

With three or more of the required AC Power Sources inoperable, the system is in a severly degraded condition and action must be taken immediately to begin an orderly shutdown in accordance with Specification 3.0.3.

L.1 and L.2

If the Required Actions cannot be met within the required Completion Times, the unit must be in MODE 3 within 6 hours and MODE 5 within the following 30 hours. These times allow for a controlled shutdown of the reactor without placing undue stress on the plant operators or plant systems.

COMPLETION TIMES

The allowable out-of-service times are based on Regulatory Guide 1.93, Rev.[ ], ["Availability of Electric Power Sources"], as modified by the diesel generator manufacturer's recommendations (Ref. 5). With only one AC power source inoperable, 72 hours is allowed to restore it to OPERABLE status. With more than one AC power source inoperable, more restrictive Completion Times are imposed to maintain an equivalent level of AC power source reliability.

Operating experience indicates that the availability of a typical offsite source is higher than that of a typical diesel generator. Thus, if risk is evaluated in terms of availability, the risk associated with the loss of an offsite power source (the source with the higher availability) would appear to be more severe than the risk associated with the loss of a diesel generator (the source with the lower availability). However, this apparent difference in severity is offset by maintainability considerations: that is, the time required to detect and restore an unavailable offsite source is generally much less than that required to detect and restore an unavailable diesel generator. Based on these considerations, a general distinction between operating restrictions associated with the loss of an offsite source and those restrictions associated with the loss of a diesel generator is not warranted.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The AC power sources are designed to permit inspection and testing of all important areas and features, especially those which have a standby function. Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9 ,"Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Ref. 2), and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Ref. 6).

To minimize the wear on moving parts which are not lubricated during engine standby conditions, all diesel generator starts for this surveillance may be preceded by engine prelube and warm-up procedures as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine are minimized. At WBN the diesel engines are continuously prelubed by an oil circulating pump and a heater. While not specifically mentioned with each SR this setup is implied for all diesel generator starts.

SR 3.8.1.1

This Surveillance Requirement assures proper circuit continuity for the offsite AC power supply to the onsite distribution network and availability of offsite AC power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. A 7-day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and its status is displayed in the Control Room.

SR 3.8.1.2

This surveillance helps to ensure the availability of the diesel generators to mitigate design basis transients and accidents and maintain the unit in safe shutdown conditions. The requirement to achieve the specified voltage and frequency supports the assumptions in the design basis LOCA analysis (Ref. 7).

The diesel generator start in [10] seconds from standby conditions need only be performed once per 184 days per the requirements of SR 3.8.1.5.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This surveillance demonstrates that the standby diesel generators are capable of synchronizing and accepting the equivalent of greater than or equal to accident loads. This allows for manual loading. The 60 minute run time for the diesel generator (required by Ref. 6, para. 2.c.(2)) is to stabilize the engine temperature. This will ensure that cooling and lubrication are adequate for extended periods of operation. All diesel engine runs for this surveillance may include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine are minimized (Ref. 8).

SR 3.8.1.4

This surveillance verifies that sufficient air capacity at the proper pressure for a minimum of [ ] engine start[s] is available without the aid of the [refill] air compressor. The system design requirement provides for [ ] engine starts (Ref. 9). However, only 1 start is assumed for all safety analyses. The required air start receiver pressure of ≥ [ ] psig is based on the air pressure required for [ ] air start receiver to provide sufficient air pressure and flow to achieve diesel generator start in [ ] seconds. The surveillance frequency is established by [ ].

SR 3.8.1.5

This surveillance demonstrates the availability of the standby power supply to mitigate design basis transients and accidents and maintain the unit in safe shutdown conditions. The [ ] second time requirement to achieve the specified voltage and frequency supports the assumptions in the design basis LOCA analysis (Ref. 7). SR 3.8.1.3 is to be performed after this surveillance.

SR 3.8.1.6

Transfer of the unit power supply from the normal circuit to the alternate circuit demonstrates the OPERABILITY of the alternate circuit distribution network to feed the shutdown loads. The 18 month surveillance frequency is consistent with refueling cycles and is performed when the unit is not in MODES 1-4.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.7

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 surveillance demonstrates the diesel generator load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency limits. For Diesel Generator 1A, the largest single load is [600] kW [essential raw cooling water pump], for Diesel Generator 1B, [600] kW [essential raw cooling water pump], for Diesel Generator 2A, [600] kW [essential raw cooling water pump], and for Diesel Generator 2B, [600] kW [essential raw cooling water pump] (Ref. 3). As required by Reference 10, paragraph 6.3.1(3), the load rejection test is acceptable if the increase in the speed of the diesel does not exceed 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower. The Surveillance frequency is consistent with the refueling cycle.

SR 3.8.1.8

This surveillance demonstrates the diesel generator capability to reject a full load without exceeding the predetermined voltage limits. The generator full load rejection may occur due to a system fault or inadvertent breaker tripping. This surveillance verifies proper engine-generator load response under the simulated test conditions. This test will simulate the total connected loads that the diesel generator will experience, following a full load rejection, and verify that the diesel generator will not trip on loss of the load. This SR is not to be performed in MODES 1-4, therefore, the frequency is consistent with the refueling cycle.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

As required by Reference 6, paragraph 2.a.(1), this surveillance demonstrates the as-designed operation of the standby power sources during loss of the offsite power source. This test verifies all actions encountered from the loss of offsite power to shedding of the nonessential loads and energizing of the ESF buses from the standby power sources. It further demonstrates the capability of the diesel generator to automatically achieve the required voltage and frequency within the specified time. This SR is not performed in MODES 1-4, therefore, the frequency is consistent with the refueling cycle.

SR 3.8.1.10

This is still an open item with NRC. WBN proposes a test involving a loss of offsite powr (LOOP) with a delayed ESF signal to ensure that the blackout (BO) load sequencing resets to the [ESF-BO] load sequencing. This would seem to be a more representative test because at WBN when a ESF signal is recieived, all loads are energized with no sequencing if power is available.

SR 3.8.1.11

This surveillance demonstrates that the standby power source diesel generator automatically starts and achieves the required voltage and frequency within the specified time from the design basis ESF actuation signals and remains on standby for ≥ [5] minutes. The [5] minute period provides sufficient time to demonstrate OPERABILITY and achieve stability. This surveillance also verifies that permanently connected and emergency loads are energized from the offsite power circuits.

SR 3.8.1.12

This surveillance demonstrates the standby power source operation as discussed in the Bases for SR 3.8.1.9 during loss of offsite power in conjunction with an [ESF] actuation signal. The [ESF] actuation signal is given after verification of load shedding.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued

SR 3.8.1.13

This surveillance demonstrates that diesel generator non-critical protective functions, e.g. [high jacket water temperature], are bypassed on an [ESF] actuation signal. The non-critical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This provides the operator with sufficient time to react appropriately. The diesel generator 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 diesel generator. Critical protective functions, such as engine overspeed and generator differential current, trip the diesel generator to avert substantial damage to the diesel generator unit.

SR 3.8.1.14 and SR. 3.8.1.15

Regulatory Guide 1.108,"Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", (Ref. 6, Para. 2.a.(3)) requires demonstration once per 18 months that the diesel generators can start and achieve the proper voltage and frequency within the specified time and also run continuously with the proper voltage and frequency. SR 3.8.1.14 will verify the load carrying capability of the diesel generator at 110 % continuous rating for 2 hours and 100 % rating for 22 hours. SR 3.8.1.15 demonstrates the functional capability of the Diesel Generator to restart within 5 minutes after being shutdown from a hot condition and achieve the proper voltage and frequency within the specified time.

SR 3.8.1.16

As required by Reference 6, paragraph 2.a.(6), this surveillance assures that the synchronization and load transfer from the diesel generator to the offsite power source can be made and the diesel generator can be returned to standby status when offsite power is restored. It also ensures that the auto-start logic is reset to allow the diesel generator to reload if a subsequent loss of offsite power occurs. The diesel generator is considered to be in standby status when its output breaker is open and it can receive a start signal and go through the loading sequence.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.17

Demonstration of the test mode override ensures that the diesel generator availability under accident conditions will not be compromised as the result of testing. Interlocks to ESF actuation and loss-of-offsite power sensing circuits cause the diesel generator to automatically reset to its normal mode if either of these signals is received during operation in the test mode.

SR 3.8.1.18

As required by Reference 6, paragraph 2.a.(2), each diesel generator is required to demonstrate proper operation for the DBA loading sequence to ensure that voltage and frequency are maintained within the required limits. Under accident conditions, prior to connecting Diesel Generators to their appropriate bus, all loads are shed except load center feeders and those motor control centers which feed safety-related loads (referred to as permanently connected loads). Upon reaching rated voltage and frequency, the diesel generators are then connected to their respective buses. Loads are then sequentially connected to the buses by the automatic sequencer. The sequencing logic controls the permissive and starting signals to motor breakers so as to prevent an overburden of the power source by automatic load application. Reference 12 provides a summary of the automatic and manual loading and unloading of ESF buses.

SR 3.8.1.19

This surveillance demonstrates that the diesel generator starting, connection, and loading interdependence has not been compromised following 10 years of operation or a modification. Also, this surveillance demonstrates that each engine is capable of achieving proper speed within the specified time when started simultaneously.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

Surveillance Frequencies

In general, surveillance frequencies are based on engineering judgement and industry accepted practice considering the plant conditions required to perform the test, the ease of performing the test and the likelihood of a change in system or component status.

Diesel Generator Test Schedule:

The diesel generator test schedule (Table 3.8.1-1) implements the recommendations of Regulatory Guide 1.9, Revision 3. The purpose of this test schedule is to provide sufficiently timely test data to establish a confidence level associated with the goal to maintain diesel generator reliability above 0.95 per demand.

Per Regulatory Guide 1.9, Revision 3, each diesel generator unit should be tested at least once every 31 days. Whenever a diesel generator has experienced 4 or more failures in the last 25 demands, the maximum time between tests is reduced to 7 days. Four failures in 25 demands is on the threshold of acceptable diesel generator performance, and hence may be an early indication of diesel generator reliability degradation. However, when considered in the light of a long history of tests, four failures in the last 25 demands may only be a statistically probable distribution of random events. Increasing the test frequency will allow for a more timely accumulation of additional test data upon which to base judgement of the reliability of the diesel generator.

(continued)

BASES (continued)

REFERENCES

1. 10CFR50, General Design Criteria 17, "Electric Power Systems".

2. Regulatory Guide 1.9, [Rev. 3], "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", [DRAFT].

3. Watts Bar FSAR, Tables [8.3-1].

4. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors", United States Nuclear Regulatory Commission, February 6, 1987".

5. Regulatory Guide 1.93, Rev. [ ], "Availability of Electric Power Sources", [,].

6. Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", [1971].

7. Watts Bar FSAR, Section [6.3.3], Table [6.3.1].

8. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability", July 2, 1984.

9. Watts Bar FSAR, Section [9.5.6.1].

10. IEEE Standard 387-[ ], "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations".

11. IEEE Standard 308-[1971], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations".

12. Watts Bar FSAR, Table [8.3-4].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources-Shutdown

BASES

BACKGROUND

The AC power system consists of the offsite power sources, (preferred power), and the onsite standby power sources. As required by 10CFR50, Appendix A, General Design Criteria 17, "Electric power systems", (Ref. 1), the design of the AC power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The safety-related station loads are divided into two trains, with each train fed by an ESF bus. Each ESF bus has two separate and independent offsite sources of power. Either of the two trains provide for the minimum safety functions necessary to shutdown the unit and maintain it in a safe shutdown condition. An electrical power distribution system diagram is provided in Figure B 3.8.1-1.

AC power from the [161] kV transformer yard to the onsite electrical distribution system is supplied by two independent circuits. The two independent circuits, supply the 6900 VAC ESF buses

The normal feeds are:

- From Unit Station Service transformer (USST) 1A through breaker 1114 to Unit Board 1B through breaker 1712 to breaker 1718 on 6.9 kV Shutdown Board 1A-A.

- From USST 1B through breaker 1122 to Unit Board 1C through breaker 1722 to breaker 1726 on 6.9 kV Shutdown Board 1B-B.

- From USST 2A through breaker 1214 to Unit Board 28 through breaker 1814 to breaker 1818 on 6.9 kV Shutdown Board 2A-A.

- From USST 2B through breaker 1222 to Unit board 2C through breaker 1822 to breaker 1826 on 6.9 kV Shutdown Board 2B-B.

The unit boards may also be manually or automatically transferred to offsite power.

(continued)

BASES (continued)

BACKGROUND (continued)

The first alternates (offsite power) are:

- From common station service transformer (CSST) C breaker 1712 to breaker 1716 to 6.9 kV Shutdown Board 1A-A.
- From CSST D breaker 2814 to breaker 1728 to 6.9 kV Shutdown Board 1B-B.
- From CSST transformer C breaker 1712 to breaker 1816 to 6.9 kV Shutdown Board 2A-A.
- From CSST D breaker 2814 to breaker 1828 to 6.9 kV Shutdown Board 2B-B.

A single offsite circuit is capable of providing the ESF loads. One of these circuits are required to meet the Limiting Condition for Operation.

The on-site standby power source for each 6900 VAC ESF bus is a diesel generator. WBN uses 4 diesel generator sets for Unit 1 operation. These same diesel generators will be shared for Unit 2 operation. WBN also utilizes a C-S diesel generator that can be aligned to any 6.9 kV ESF bus. The diesel generator starts automatically on safety injection signal, or a loss of offsite power, e.g. bus loss of voltage.

Ratings for train 1A, 1B, 2A, and 2B diesel generators, satisfy the requirements of Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", (Ref. 2). The continuous service rating of each of the diesel generators is [4400] kW.Each diesel generator set also has an additional rating of 4840 kW for 2 hours out of 24. The ESF loads which are powered from the [6.9] kV emergency buses are listed in Reference 3.

The plant [6.9] kV buses provide AC electrical power for the safety related equipment. A decrease or loss of offsite power on the bus is detected by degraded and loss of voltage for initiation of the load shedding and sequencing logic which results in offsite power circuit breaker trip, diesel generator start and load sequencing. The under voltage relays sense loss of voltage to a level at which electrical equipment would not function. The degraded voltage relay trip when the voltage condition is such that continuous operation of the equipment is undesirable.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

A reduced compliment of AC Sources is adequate in MODES 5 and 6 to assure power for systems required to recover from postulated events in these MODES, e.g. a fuel handling accident (Ref. 3).

The AC Power System is part of the primary success path which functions or actuates to mitigate a MODE 5 or 6 Design Basis Accident (DBA) or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 4).

LCOs

One circuit between the offsite transmission network and the safety-related distribution system, and two diesel generators (either 1A-A and 2A-A or 1B-B or 2B-B) are required to be OPERABLE to ensure the availability of the required power to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel, e.g. fuel handling accident.

APPLICABILITY

The OPERABILITY of AC Sources in MODES 5 and 6 is required to assure: (1) adequate coolant inventory for the irradiated fuel in the core, (2) recovery from a fuel handling accident, (3) sufficient power for support systems required for normal operations, e.g. decay heat removal, refueling activities, component cooling, and, (4) sufficient instrumentation and control capability for monitoring and maintaining the unit status.

AC power requirements for MODE 1, 2, 3, and 4 are covered in LCO 3.8.1, AC Sources-Operating.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, A.3, A.4, A.5, A.6.1 and A.6.2

------------------------NOTE------------------------------

Suspension of Required Actions shall not preclude completion of actions to establish a safe conservative condition.

With one or more required AC Power Sources inoperable, it is required to suspend CORE ALTERATIONS, to suspend the movement of irradiated fuel, to suspend the movement of loads over irradiated fuel, to suspend operations involving positive reactivity additions, and to suspend any activities which could potentially result in inadvertent draining of the reactor vessel. These actions will preclude the occurrence of actions which could potentially initiate the postulated events. It is further required to initiate action to restore the required AC electrical power sources within 15 minutes and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit's safety systems.

The Completion Time of 15 minutes is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time the unit's safety systems may be without power.

SURVEILLANCE REQUIREMENTS

The following Surveillance Requirements are applicable to these Bases.

SR 3.8.1.1 through SR 3.8.1.6;
SR 3.8.1.8 through SR 3.8.1.11;
SR 3.8.1.15, SR 3.8.1.16, and
SR 3.8.1.18

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A General Design Criteria for Nuclear Power Plants, No. 17, " Electric power systems".

(continued)

BASES (continued)

REFERENCES (continued)

2. Regulatory Guide 1.9, Rev. [3], "Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", [DRAFT].

3. Watts Bar FSAR, Section [15.7.4].

4. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987".

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel and Lubricating Oil

BASES

BACKGROUND

For the purpose of this LCO, the diesel fuel oil subsystem is considered to include 1) fuel oil storage, 2) fuel oil transfer capabilities, and 3) fuel oil properties.

Each diesel generator is provided with a storage tank having a fuel capacity sufficient to operate that diesel for a period of [7] days while the diesel generator is supplying maximum post-LOCA load demand (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any two diesel generators is available. This onsite fuel capacity lasts longer than the time it would take to replenish the onsite supply from outside sources.

A 550 gallon day tank is provided for each diesel engine. each day tank is housed in a separate room in the diesel generator building and has fuel capacity for approximately 2 hours of full-load operations (Ref. 1). Fuel oil is tranferred from storage tanks to the day tanks by either of two pumps located on each storage tank. 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 diesel generator. In the event that the piping between the last isolation valve and the day tank breaks, the use of one diesel can be lost. This occurs only after the two hour supply of fuel in the day tank has been used. All of the tanks, pumps, and piping are underground.

During operation of the diesel generators, fuel oil pumps driven by the diesel engines tranfer fuel from the day tanks to the diesel engine fuel manifolds. Level controls mounted on the day tanks automatically start and stop the storage tank transfer pumps.

In addition, alarms both locally and in the control room annunciate low level and high level in any day tank.

In the unlikely event of a failure in one of the supply trains, the associated day tank low-level alarm annunciates when the fuel oil remaining in the tank provides approximately 2 hours of full-load operation, thus allowing the operator to take corrective action to prevent the loss of the diesel.

(continued)

BASES (continued)

BACKGROUND	For proper operation of the standby diesel generators, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guid 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 3). The fuel oil properties governed by these surveillance requirements are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level. (Discussion of lubricating oil to be provided later).
APPLICABLE SAFETY ANALYSES	The initial conditions of design basis transient and accident analyses in the FSAR Chapters 6, Engineered Safety Features, and 15, Accident Analyses, assume all ESF systems are OPERABLE. The diesel fuel oil subsystem and lubricating oil provide the necessary supply to support operation of the diesel generators. Diesel Fuel and Lubrication Oil satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements since it supports the operation of the standby AC power sources.
LCOs	The diesel fuel oil subsystem is required to be OPERABLE and sufficient lubricating oil supply available to ensure availability of the required AC power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident. This requires that the associated day tank and storage tank for each diesel generator be OPERABLE, that the fuel oil properties be within acceptable limits and that ≥ (930) gallons of lubricating oil be available onsite.
APPLICABILITY	The diesel fuel oil subsystem is required to be OPERABLE and sufficient lubricating oil supply is required to be available when the associated diesel generator is required to be OPERABLE (Refer to LCO 3.8.1 and LCO 3.8.2).

(continued)

BASES (continued)

ACTIONS

A.1

With the fuel oil level in the day tank low, insufficient fuel oil is available to satisfy the design basis requirement of supporting two hours of continuous operation. Consequently, a short time (one hour) is allowed to restore the fuel level. This one hour Completion Time is acceptable given the low probability of a diesel start requirement this period and the automatic start of the storage tank transfer pumps on low-level. If the day tank level is below (250) gallons, the diesel generator must be declared inoperable and the Required Actions of LCO 3.8.1 or LCO 3.8.2, as applicable, initiated.

B.1

With the fuel oil transfer capability inoperable, sufficient fuel oil is available in the day tank to support 2 hours of continuous diesel generator operation. If the tranfer capability cannot be restored within this interval, the diesel generator must be declared inoperable.

C.1

Low level in one or more fuel storage tanks indicates that the design basis requirement of supporting seven days of continuous operation may not be able to be satisified. Consequently, a period of 24 hours is allowed to restore the fuel level. This 24 hours Completion Time is acceptable given the fuel oil transfer capability between storage tanks and the high likelihood that additional fuel could be provided from offsite. If the storage tank level is below (60,000) gallons, the associated diesel generator must be declared inoperable and the Required Actions of LCO 3.8.1 or LCO 3.8.2, as applicable, initiated.

D.1

With lubricating inventory less than [331] gallons per diesel engine, sufficient lubricating oil to support 7 days of continuous diesel generator operation at full load conditions may not be available. However, a limited time (24 hours) is provided to restore the lubricating oil inventory because of the relatively low rate of usage and the high likelihood that additional supplies can be provided from offsite.

(continued)

BASES (continued)

ACTIONS

E.1

With the fuel oil properties defined in the Bases for SR 3.8.3.4, SR 3.8.3.5, and SR 3.8.3.6 not within the required limits, a period of 72 hours is allowed to restore the fuel oil properties. This provides sufficient time to retest the fuel oil sample to confirm that the initial test results were valid. Even if a diesel generator start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the diesel generator would still be capable of performing its intended function.

F.1

For the accelerated stability testing portion of the diesel fuel oil testing program, a period of 30 days is provided to restore the fuel oil properties to within limits of ANSI-N195. This test measures stability of distillate fuels under accelerated oxidizing conditions. It is performed in an extreme testing environment and results only in an index of long term storage stability. If all other typical parameters tested (as specified in SR 3.8.3.4, SR 3.8.3.5, and SR 3.8.3.6) are within limits of ASTM, fuel oil exceeding the insolubility limits of ANSI N195 Appendix B will remain stable for a period ranging from months to years: therefore, the 30-day allowable out-of-service time is acceptable.

G.1

With the Required Actions and associated Completion Times not met, or the diesel fuel oil subsystem is inoperable for reasons other than addressed by Conditions A through H, the associated diesel generator must be declared inoperable and the Actions of LCO 3.8.1 or LCO 3.8.2 followed.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

The verification of each diesel engine fuel tank supply ensures that enough fuel is on hand at the engine to sustain at least [2] hours of full load operation for the Diesel Generators (Ref 1). The 24 hour frequency is adequate to ensure a sufficient fuel supply is available since operators should be aware of large uses during this period and low-level alarms are provided.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.2

The verification of the 7-day fuel storage tank supply demonstrates that sufficient fuel is available to [sustain the full load diesel generator operation for [7] days (Ref. 1).] This time period is sufficient to put the unit in a safe shutdown condition and provides enough time to bring in replenishment fuel from an offsite location.

SR 3.8.3.3

This surveillance ensures that sufficient lubricating oil inventory is avaiable to support at least seven days of full load operation for the diesel generator. The (331) gallons per engine requirement is based on the diesel generators' manufacturers comsumption valvues for the run-time of the diesel. A 31-day frequency is adequate to ensure a sufficient lubricating oil supply is onsite since diesel generator starts and run-times are closely monitored by the plant staff.

SR 3.8.3.4, 3.8.3.5 and 3.8.3.6

For proper operation of the standby diesel generators, it is necessary to ensure the proper quality of the fuel oil. Reference 3 addresses the recommended fuel oil practices as supplemented by Reference 2, Section c.2.

Periodically and upon replenishment of the fuel oil, this surveillance is performed to establish that the diesel engine fuel meets the NRC guidelines on kinematic viscosity, water and sediment content, specific gravity, and sediment impurity (Ref. 2). The samples must be obtained in accordance with ASTM-D270 [ ] (Ref. 2, para. c.2.c). The water and sediment content and kinematic viscosity must be determined by testing in accordance with ASTM-D975-77 (Ref. 2, para. c.2.a) and the impurity level must be determined in accordance with ASTM-D2274-[ ]. The required surveillance frequencies for sampling the fuel oil are established by Reference 2, paragraph 2.b.

Typical fuel oil sample requirements are:

A. Water and sediment content ≤ [0.05]% by volume.

B. Kinematic viscosity at 40°C ≥[1.9] and ≤[4.1] centistokes.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4, SR 3.8.3.5, and SR 3.8.3.6 (continued)

C. API Gravity within [0.3] degrees at [60]°F or a specific gravity of within [.0016] at [60/60]°F.

D. A flash point ≥ [125]°F.

E. A clear and bright appearance with proper color.

F. Impurity level < [10] mg of insoluble per [1] liter.

SR 3.8.3.7 and SR 3.8.3.8

These surveillances are performed to ensure removal of any accumulated water in the fuel oil supply due to condensation or water in-leakage in the tanks and supply system. This is necessary to ensure that the fuel oil quality is within manufacturer's recommended standards. The surveillance frequencies are established by Ref. 2, paragraphs 2.d and e.

SR 3.8.3.9

This Surveillance Requirement demonstrates that the fuel oil transfer system operates and transfers fuel from 7-day storage tanks to each day tank. This is required to support the [7-day] continuous operation of the diesel generator. This surveillance provides assurance that there is no crud buildup blocking the fuel supply line and that the fuel supply line is intact.

SR 3.8.3.10

The draining of the fuel-oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning is required at 10 year intervals (Ref. 2, para. 2.f). To preclude the introduction of surfactants in the fuel system, the cleaning should be accomplished using sodium hypochlorite solutions or their equivalent rather than soap or detergents.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Chapter [9.5].
2. Regulatory Guide 1.137, Rev. [ ] "Fuel Oil systems for Standby Diesel Generators," [ ].
3. ANSI N195-[ ], "Fuel Oil Systems for Standby Diesel Generators."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources-Operating

BASES

BACKGROUND	The station DC power system provides the AC emergency power system with required control power. It also provides both motive and control power to selected safety-related equipment. The DC subsystems conform to the independence and redundancy requirements of Regulatory Guide 1.6, Rev. [0], Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems, and IEEE Standard 308-[1971], IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, (References 1 and 2).
	The DC systems consist of [4] independent and redundant safety related DC subsystems. Each of the [ ] safety related 125 volt DC power subsystems, provides the control power for its associated safety related AC power load group, [6900] kV switchgear, and [480] volt load centers. Also, these DC subsystems provide DC power to the Engineered Safety Features (ESF) valve actuation, plant alarm and indication circuits, emergency lighting system, and the DC control power for each diesel generator.. Loss of any one of the DC subsystems does not prevent the minimum safety function from being performed (Ref. 3).
	Each 125 volt DC battery is separately housed in a ventilated room apart from its charger and distribution center. Each subsystem is located in an area separated physically and electrically from other subsystems to ensure that a single failure in one train does not cause a failure in the redundant train. There is no sharing between redundant safety related trains of equipment such as batteries, battery chargers, or distribution panels (Ref. 4). WBN also has a fifth vital battery that can serve as a temporary substitute for any of the four 125 volt DC votal batteries.
	125 volt DC battery banks 1, 2, 3, and 4 each have sufficient stored energy to operate connected essential loads continuously for at least [ ] minutes. Battery size is based on [ ]% of required capacity and, after selection

(continued)

BASES (continued)

BACKGROUND (continued)	of an available commercial battery, results in a battery capacity in excess of [ ]% of required capacity (Ref. 5). Batteries are sized to produce required capacity at [ ] percent of nameplate rating, corresponding to warranted capacity at end-of-life-cycles and the 100 percent design demand. Each battery charger has adequate power-output capacity for steady-state operation of connected loads required during normal operation while maintaining its battery in a fully charged state. Each battery-charger has sufficient capacity to restore the battery from the design minimum charge to its fully charged state in $\leq$ [40] hours while supplying normal steady-state loads (Ref. 5). One battery charger is provided for each 125 volt DC bus.
APPLICABLE SAFETY ANALYSES	The initial conditions of design basis transient and accident analyses in FSAR Chapter [6], Engineering Safety Features, and Chapter [15], Accident Analyses, assume all ESF systems are OPERABLE. The DC power systems provide normal and emergency DC power for emergency auxiliaries and for control and switching during all modes of operation. The DC Sources are a system that is part of the primary success path which functions or actuates to mitigate a MODE 1, 2, 3, or 4 Design Basis Accident (DBA) or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 6).
LCOs	The DC electrical power sources are required to be OPERABLE to ensure availability of the required power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The DC electrical power sources are considered OPERABLE if the 125 volt batteries and their associated full-capacity chargers satisfy the applicable surveillance requirements.

(continued)

BASES (continued)

APPLICABILITY

The DC power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and

2. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

DC power requirements for MODES 5 and 6 are covered in LCO 3.8.5.

ACTIONS

A.1

With one of the required batteries or full capacity chargers inoperable, the available DC power supplies do not have the required redundancy; however, the DC power system has the capacity to support a safe shutdown and to mitigate an accident condition. Continued power operation should not exceed 2 hours (Section C.5 of Ref. 7).

The Completion Times are consistent with the allowable out-of-service times based upon Regulatory Guide 1.93, Rev.[ ], Availability of Electric Power Sources (Ref. 7), or industry-accepted practice and engineering judgement considering the number of available systems and the time required to reasonably complete the Required Actions.

B.1 AND B.2

If the inoperable DC Power Source cannot be restored in the required Completion Time, a controlled shutdown is required. The plant must be placed in MODE 3 within the 6 hours and in MODE 5 within the following 30 hours. These times allow for a controlled plant shutdown without placing undue stress on plant systems or operators.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of References 8 and 9.

SR 3.8.4.1

Verifying total battery terminal voltage on float charge, ensures the effectiveness of the charging system.

SR 3.8.4.2 through SR 3.8.4.5

Visual inspections of the battery cells and connections and the resistance measurement of each cell and terminal connection provides an indication of physical damage or abnormal deterioration which could potentially degrade battery performance.

SR 3.8.4.6

Paragraph C.1.b of Reference 10 requires that the battery charger supply be based on the largest combined demands of the various steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during which these demands occur. The minimum required amperes and duration for the battery charge ensure that the battery load requirements, [after the   minute,] can be satisfied (Refer to SR 3.8.4.7).

SR 3.8.4.7

Paragraph C.1.c of Reference 10 requires the performance of a battery service test at intervals not to exceed 18 months. A battery service test is a special capacity test to demonstrate the capability of the battery to meet the system design requirements. [Reference 11 provides the load requirements for the Load Group 1, 2, 3, and 4 batteries.]

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8

IEEE-450 (Ref. 10), paragraph 4.2(3) recommends a performance test for each battery at 60-month intervals. A battery performance test is a capacity test of the battery in the "as found" condition, after being in service, to detect any change in the capacity as determined by the new battery acceptance test. IEEE-450 (Ref. 10), paragraph 4.2(3) recommends [annual] performance tests of battery capacity for any battery that shows signs of degradation or has reached 85 percent of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10 % of rated capacity from its average on previous performance tests, or is below [90% of the manufacturer's rating]. However, this test cannot be performed at other than shutdown conditions. Therefore, an 18-month interval is appropriate.

In accordance with Paragraph 6 of Reference 10, the battery should be replaced if its capacity is below [80% of the manufacturer's rating]. A capacity of 80% shows the battery rate of deterioration is increasing even if there is ample capacity to meet the load requirements.

Surveillance frequencies are based on engineering judgement and industry-accepted practice considering the unit conditions required to perform the test, the ease of performing the test and the likelihood of a change in the system component status.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.6, Rev.[ ] "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems", [ ].

2. IEEE Standard 308-[ ], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations".

3. Watts Bar FSAR, Section [8.3.1.4].

4. Watts Bar FSAR, Section [8.3.2.1.3].

5. Watts Bar FSAR, Section [8.3.2.1.2].

6. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987".

7. Regulatory Guide 1.93, Rev.[ ], "Availability of Electric Power Sources", [ ].

8. Regulatory Guide 1.129, Rev. [ ], "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", [ ].

9. IEEE Standard 450-[ ], "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems".

10. Regulatory Guide 1.32, Rev. [ ]"Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants", [ ].

11. Watts Bar FSAR, Tables [8.3-6 to 8.3-8].

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources-Shutdown

BASES

BACKGROUND	A description of the DC power sources is provided in the Bases for LCO 3.2.4.
APPLICABLE SAFETY ANALYSES	A reduced compliment of DC Sources in MODES 5 and 6 is adequate to assure sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and to assure adequate power for systems and control logic required to recover from a postulated fuel handling accident (Ref. 1). The DC Sources are a system that is part of the primary success path which functions or actuates to mitigate a MODE 5 or 6 Design Basis Accident (DBA) or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 6).
LCOs	One train of the DC power sources is required to be OPERABLE in MODES 5 and 6. Either train A or train B of the DC power sources will ensure the availability of the required power to recover from postulated events in MODES 5 and 6 and when handling irradiated fuel (e.g., fuel handling accident). The DC electrical power sources are considered OPERABLE if the vital battery bank [1 and 3] or the vital battery bank [2 and 4] 125 volt batteries and their respective battery chargers satisfy the applicable surveillance requirements.
APPLICABILITY	The OPERABILITY of DC Sources in MODES 5 and 6, when moving irradiated fuel and when moving heavy loads over irradiated fuel is required to assure (1) adequate coolant inventory for the irradiated fuel in the core in case of an inadvertent draindown event, (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and (3) adequate decay heat removal.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, A.3, A.4, A.5, A.6.1, and A.6.2

-----------------------NOTE-----------------------
Suspension of these activities shall not preclude completion of actions to establish to a safe conservative condition.

With the required battery banks or associated battery chargers inoperable, DC power is not available to support operation of the Emergency Core Cooling Systems (ECCS), decay heat removal equipment, uninterrupted power supplies and switchgear required to recover from a fuel handling accident or other events postulated to occur in MODE 5 or 6. Actions are required to be taken within 15 minutes to suspend CORE ALTERATIONS, to suspend the movement of irradiated fuel, to suspend the movement of loads over irradiated fuel, to suspend operations involving positive reactivity additions, and to suspend any activities which could potentially result in the inadvertent draining of the reactor vessel. These actions will preclude the occurrence of actions which could potentially initiate the postulated events. It is further required to initiate action to restore one DC power train within 15 minutes and to continue this action until restoration is accomplished in order to provide the necessary DC power for the unit's safety systems.

The Completion Time of 15 minutes is consistent with the required times for actions requiring prompt attention. The restoration of one DC power train should be completed as quickly as possible in order to minimize the time the unit's safety systems may be without power.

SURVEILLANCE REQUIREMENTS

The Bases provided for SR 3.8.4.1 through 3.8.4.8 are applicable.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.6, Rev. [0], "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems", [ ].

2. IEEE Standard 308-[ ], "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations",

3. Watts Bar FSAR, Section [8.3.1.4].

4 Watts Bar FSAR, Section [8.3.2.1.3].

5. Watts Bar FSAR, Section [8.3.2.1.2].

6. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

Table 3.8.6-1 specifies the limits for three different categories. The categories are identified as:

A - Limits for each designated pilot cell,
B - Limits for each connected cell, and
C - Allowable value for each connected cell.

Parameters are identified for each category. These parameters are:

1. Electrolyte level
2. Float voltage, and
3. Specify Gravity (sg).

The limits for the designated pilot cell's float voltage and specifc gravity (greater than or equal of [2.13] volts and 1.200 (sg) or a battery charger current current that had stabilized at a low value) are based upon the manufacturer's recommended values (Ref. 1), and are characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity (greater than [2.13] volts and [1.195] (sg), and an average specific gravity of all the connected cells greater than or equal to [1.205] (sg)) are the values recommended by the manufacturer (Ref. 1) to ensure the OPERABILITY and capability of the battery.

The specific gravity readings must be corrected for electrolyte temperature. The specific gravity readings must also be corrected for electrolyte level unless the charging current is less than [ ] ampere for the [vital] batteries (whichever is applicable) and] the electrolyte level is within the high- and low-level limits specified in Table 3.8.6-1 during accident conditions, (2) an assumed loss of off-site power, and (3) a single failure of the other standby A.C. source.

Battery Cell Parameters satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement on Technical Specification Improvements as documented in Reference 2.

(continued)

BASES (continued)

LCO

The vital batteries [and diesel generator] D.C. power sources are required to be OPERABLE to ensure availability of the required power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated design basis accident. The batteries are considered OPERABLE if the Category A and B limits are met.

APPLICIBILITY

The battery electrolyte must be within the limits of Table 3.8.6-1 when the associated D.C. power sources [and diesel generators] are required to be OPERABLE.

ACTIONS

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

Operation with one or more cells in one or more batteries' parameters outside the normal limit (i.e., Category A limits not met or Category B limits not met or Category A and B limits not met) but within the allowable value (Category C) specified in Table 3.8.6-1 is permitted since sufficient capacity exists to perform the intended function. The pilot cells electrolyte level and float voltage are required to be demonstrated to meet the Category C allowable values within 1 hour. This check will provide a gross indication of the status of the remainder of the battery's cells since the pilot cells are typically the worst cells. One hour provides time to inspect the electroylte level and to confirm the float voltage of the [4] pilot cells.

The requirement to demonstrate the Category C allowable values are met within 24 hours will ensure that during the time to restore the parameters to the Category A and B limits that the battery will still be capable of performing its intended function. Twenty-four hours are provided to complete this Required Action because specific gravity measurements must be obtained for each connected cell.

(continued)

BASES (continued)

REFERENCES

1. [Battery Manufacturer recommended parameter values].

2. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

3. IEEE Std. 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

4. IEEE Std. 308-1978, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Distribution System - Operating

BASES

BACKGROUND

The AC onsite power system is comprised of 4 redundant and independent [6.9] kV Engineered Safety Features (ESF) distribution systems with their [480] volt load centers and motor control centers, 120 volt vital AC power system and the standby power supplies (diesel generator units). Each [6.9] kV ESF bus is normally connected to a preferred source which is one of the two independent offsite sources of [161] kV power. During a loss of one offsite power source to the [6.9] kV ESF buses, a [161] kV transfer scheme is accomplished by utilizing a time delayed bus under voltage relay. If all offsite sources are unavailable, the onsite emergency power system will supply power to the [6.9] kV ESF buses. Refer to Figure B 3.8.1-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) transients and accident analyses in FSAR Chapters [6], Engineered Safety Features (ESF), and [15], Accident Analyses, assume OPERABLE ESF systems. The electrical power distribution systems are designed to provide sufficient capacity, redundancy and reliability to ensure the availability of the necessary power to ESF systems so that the fuel, reactor coolant system, and containment design limits are not exceeded.

The availability of the power sources is consistent with the initial assumptions of the accident analyses and are based upon maintaining at least one of the onsite AC and DC power source trains and associated distribution system trains OPERABLE during and following accident conditions, assuming a loss of offsite power, and a single active failure.

The Power Distribution System is part of the primary success path which functions or actuates to mitigate a MODE 1, 2, 3, or 4 Design Basis Accident (DBA) or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

(continued)

BASES (continued)

LCOs

The power distribution system trains listed in Table 3.8.7-1 ensure the availability of the required power to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated Design Basis Accident (DBA).

For the purpose of performing an equalizing charge on two battery banks, two inverters may be disconnected from their associated DC Buses for ≤ 24 hours. During this period, the associated AC Vital Buses must be energized from the constant voltage source and the other AC Vital Buses must be energized from their associated inverters connected to their associated DC Buses.

APPLICABILITY

The power distribution systems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

1. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences or abnormal transients, and

2. Adequate core cooling is provided and containment integrity and other vital functions are maintained in the event of a postulated DBA.

Power distribution system requirements for MODES 5 and 6 are covered in the bases for LCO 3.8.8.

ACTIONS

A.1

With one or more DC Buses in one train inoperable, the remaining DC Bus distribution system train is capable of supporting the minimum safety functions necessary to shutdown the reactor and maintain it in a safe shutdown condition. However, an additional single failure could result in insufficient DC power for the minimum required ESF functions. Therefore, the inoperable DC distribution system train must be returned to OPERABLE status within 2 hours or a controlled plant shutdown must be performed per Required Actions C.1 and C.2.

(continued)

BASES (continued)

ACTIONS (continued)

A.2

With one or more AC Buses (except AC vital busses) in one train inoperable, the remaining OPERABLE Emergency Bus distribution system train is capable of supporting the minimum safety functions necessary to shutdown the unit and maintain it in the safe shutdown condition. However, an additional single failure could result in insufficient power for the minimum required ESF functions. Therefore, the inoperable distribution system train must be restored to OPERABLE status within 8 hours, or a controlled shutdown must be performed per Required Actions C.1 and C.2.

B.1 and B.2

With one of the AC Vital Buses in either distribution system train not energized from its associated inverter and associated DC Source, the Bus must be re-energized within 2 hours. The constant voltage source powered from the AC Emergency Bus is normally used to re-energize the AC Vital Bus. If the AC Vital Bus cannot be restored to OPERABLE status within the 24 hour time period, a controlled plant shutdown must be performed per Required Actions C.1 and C.2.

C.1 and C.2

With a Required Action not met within the required Completion Time, Condition D requires the plant to be placed in MODE 3 within 6 hours and in MODE 5 within the following 30 hours. The Completion Time for this action is considered by engineering judgement to be appropriate to complete this action in a safe and orderly manner.

Completion Times

The allowable Completion Times are based on industry accepted practice and engineering judgement considering the number of available systems, the importance of the systems to plant safe operations and the time required to reasonably complete the actions.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This surveillance demonstrates that the electrical power distribution system is functioning properly with all the desired circuit breakers closed and the buses energized. The verification of proper voltage availability on the buses of both trains ensures that the required power is readily available for motive and control functions for the critical system loads connected to these buses.

The frequency of 7 days is based on engineering judgement. This frequency has been shown to be acceptable through operating experience.

REFERENCES

1. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems-Shutdown

BASES

BACKGROUND	A description of the electrical power distribution system is provided in the Bases for LCO 3.8.7.
APPLICABLE SAFETY ANALYSES	The number of distribution systems necessary in MODES 5 and 6 are less than required for MODES 1 through 4. Sufficient power and distribution systems in MODES 5 and 6 are needed only to maintain the plant in the shutdown or refueling condition, provide sufficient instrumentation and control capability to monitor and maintain plant status, and assure adequate power for recovery from a postulated fuel handling accident (Ref. 1). The Power Distribution System is part of the primary success path which functions or actuates to mitigate a MODE 5 or 6 Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such it satisfies the requirements of Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 2).
LCOs	One energized power distribution system train identified in Table 3.8.7-1, consisting of the AC Emergency Buses, DC Buses, and AC Vital Buses, ensures the availability of the required power to recover from postulated events in MODES 5 and 6 (e.g., fuel handling accident [or boron dilution accident]).
APPLICABILITY	The OPERABILITY of the electrical distribution systems in MODES 5 and 6 is required to assure: (1) adequate coolant inventory for the irradiated fuel in the core ; (2) recovery from a fuel handling accident; (3) sufficient power for support systems required for normal operations (e.g., decay heat removal, refueling activities, component cooling); and (4) sufficient instrumentation and control capability for monitoring and maintaining the unit status.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, A.3, A.4, A.5, A.6.1 and A.6.2

---NOTE---
Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

With no power distribution system train energized, insufficient power is available to recover from a postulated fuel handling accident or other events in MODE 5 or 6. Consequently, it is required within 15 minutes to suspend CORE ALTERATIONS, to suspend the movement of irradiated fuel, to suspend the movement of loads over irradiated fuel, suspend operations involving reactivity additions, and to suspend any activities which could potentially result in the inadvertent draining of the reactor vessel. These actions will preclude the occurrence of actions which could potentially initiate the postulated events. It is further required to initiate action to restore one power distribution system train within 15 minutes and to continue this action until restoration is accomplished in order to provide the necessary power to the unit's safety systems.

The Completion Time of 15 minutes is consistent with the required times for actions requiring prompt attention. The restoration of one power distribution system train should be completed as quickly as possible in order to minimize the time the unit's safety systems may be without power.

SURVEILLANCE REQUIREMENTS

The Bases provided for SR 3.8.7.1 is applicable.

REFERENCES

1. Watts Bar FSAR, Section [15.7.4].

2. NRC Interim Policy Statement, "52FR3788, Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987."

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentration of the Reactor Coolant System (RCS), refueling cavity and refueling canal during refueling ensures that the reactor will remain subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the reactor coolant during refueling or fuel handling. The soluble boron concentration will offset the fuel reactivity and is measured by chemical analysis of the reactor coolant.

General Design Criteria 26 of 10 CFR Part 50, Appendix A requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The unit is brought to shutdown conditions before beginning operations to open the RCS for refueling and while the reactor coolant chemistry is under administrative control from the makeup system. The makeup system consists of the Boric Acid Transfer and Charging subsystems of the CVCS. After the unit is cooled and depressurized, and the reactor vessel head is unbolted, the head is slowly raised, and the refueling cavity and canal are flooded by pumping water from the Refueling Water Storage Tank (RWST) through the open reactor vessel using the Residual Heat Removal (RHR) System pumps or the Refueling Water Purification System.

If additions of boron are required after the vessel has been opened, the CVCS makes the additions through the RCS and open vessel. The pumping action of the RHR System and natural circulation due to thermal driving heads in the vessel and cavity are relied upon to mix the added concentrated boric acid with the water in the RCS and the refueling canal. The RHR System is kept in service during the refueling period to remove core decay heat and to provide forced circulation in the RCS. The increased concentration of boron enters the RCS cold leg charging connection and goes directly to the core.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The refueling boron concentration is consistent with the initial conditions assumed in the boron dilution accident safety analysis and is conservative for MODE 6. The boron concentration value specified in the CORE OPERATING LIMITS REPORT (COLR) includes a conservative uncertainty allowance. The required boron concentration is based on the nuclear design of each fuel cycle. This required boron concentration and the unit refueling procedures, which verify the correct fuel loading plan (including full core mapping, ensure that the k_{eff} of the core will be ≤ 0.95 during the refueling operation.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling cavity, the refueling canal, and the reactor vessel form a single mass. As a result, the soluble boron concentration is the same in each (Ref. 2).

The RCS boron concentration in MODE 6 is a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier for the units which analyze the MODE 6 boron dilution accident. For these units, the boron concentration in MODE 6 satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 3).

LCOs

The LCO requires that a minimum boron concentration be maintained while in MODE 6. The boron concentration limit ensures that a $k_{eff} \leq 0.95$ is maintained during fuel handling operations. Violation of the LCO could lead to possible inadvertent criticality during MODE 6.

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_{eff} of ≤ 0.95. Above MODE 6, LCO 3.1.1, Shutdown Margin, ensures that an adequate amount of negative reactivity is available to shutdown the reactor and to maintain the reactor subcritical.

(continued)

BASES (continued)

ACTIONS

-------------------------NOTE---------------------------
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

A.1

Continuation of CORE ALTERATIONS or positive reactivity additions is contingent upon maintaining the unit in compliance with the LCO. With the boron concentration of any of the filled portions of the RCS, the refueling canal, or the refueling cavity less than the limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended. The 15 minute Completion Time ensures prompt action is taken.

A.2

In addition to A.1, boration to restore the concentration must be initiated. The Completion Time of 15 minutes is the time allowed for an operator to correctly align and start the required systems and components. Once boration is initiated, it must be continued until the boron concentration is restored.

In the determination of the required combination of boration flow rate and boron concentration, there is not a unique design basis event which 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 of the RCS as soon as possible, the boration solution should be a highly concentrated solution of boric acid.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This surveillance verifies the reactor coolant boron concentration is within its COLR limit and is performed prior to entering MODE 6. This surveillance also ensures that the required refueling cavity boron concentration is maintained during refueling operations. The boron concentration in the coolant is determined periodically by chemical analysis. Care must be taken to obtain a representative sample. Because the likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote, a minimum frequency of once every 72 hours is reasonable. The surveillance interval is based on engineering judgment and industry operating experience and ensures that the boron concentration is checked at adequate intervals.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, Section III, Criterion 26, "Reactivity Control System Redundancy and Capability."

2. NS-57.2, ANSI/ANS-57.2-1983, Section 6.4.2.2.3, American Nuclear Society, American National Standard, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," 1983.

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves

BASES

BACKGROUND

To prevent inadvertent dilution of the reactor coolant during MODE 6 operations, all isolation valves for reactor makeup water sources containing unborated water which are connected to the Reactor Coolant System (RCS) shall be closed. The isolation valves must be secured in the closed position.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event occurring during refueling operations is precluded by adherence to this LCO which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents flow of unborated water to the filled portion of the RCS.

The valves are used to isolate unborated water sources. These valves have the potential to, indirectly, allow dilution of the RCS boron concentration in MODE 6. RCS boron concentration in MODE 6 is a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. This LCO is included because boron concentration satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 1).

LCOs

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent inadvertent boron dilution during MODE 6. Violation of the LCO could lead to inadvertent criticality during MODE 6.

These valves are permitted to be open when required for planned dilution or makeup activities in MODE 6. This allows required activities to be performed under administrative control when CORE ALTERATIONS are not being conducted.

(continued)

BASES (continued)

APPLICABILITY

An inadvertent boron dilution event in MODE 6 is prevented by this LCO which ensures isolation of all sources of unborated water to the RCS. The boron dilution accident is analyzed for all other MODES and is capable of being mitigated.

ACTIONS

-------------------------NOTE---------------------------
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

<u>A.1</u>

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with the LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be halted immediately.

<u>A.2</u>

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. Actions must be initated immediately to secure the valve in the closed position.

<u>A.3</u>

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 is performed to verify that the required boron concentration exists. The Completion Time of 4 hours is based on the practical amount of time necessary to perform the surveillance.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

One of the following valve combinations shall be verified locked closed or secured in position. Because the likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote, a minimum frequency of once every 31 days is reasonable. The surveillance interval is based on engineering judgment and industry operating experience and ensures that valve positions and lockouts are checked at adequate intervals. In addition, the requirement to perform this surveillance once within 1 hour following a planned dilution or makeup activity ensures that the valves are not inadvertently left open or unsecured.

Combination A	Combination B	Combination C	Combination D
a. 1-81-533	a. 1-81-533	a. 1-81-533	a. 1-81-533
b. 1-62-922	b. 1.62-922	b. 1-62-907	b. 1-62-907
c. 1-62-916	c. 1-62-916	c. 1-62-914	c. 1-62-914
d. 1-62-933	d. 1-62-940	d. 1-62-921	d. 1-62-921
	e. 1-62-696	e. 1-62-933	e. 1-62-940
	f. 1-62-929		f. 1-62-929
	g. 1-62-932		g. 1-62-932
	h. 1-FCV-62-128		h. 1-62-696
			i. 1-FCV-62-128

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The installed source range monitors 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 use of portable detectors is permitted, provided the LCO requirements are met.

The installed source range monitors are fission chambers. The detectors monitor the neutron flux in counts per second (cps) and cover the full range of neutron flux. If used, portable detectors should be functionally equivalent to the installed NIS source range monitors but only cover 6 decades of neutron flux.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the source range neutron flux monitors is required to provide a signal to alert the operator to changes in core reactivity such as a boron dilution accident or an improperly loaded fuel assembly. The source range neutron flux monitors are part of the primary success path which functions or actuates to 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. As such, these monitors satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 1).

LCOs

The OPERABILITY of two source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in core reactivity.

APPLICABILITY

The source range neutron flux monitors must be OPERABLE in MODE 6 to determine changes in core reactivity. No other direct means are available. The installed source range detectors and circuitry are also required to be OPERABLE in MODES 2, 3, 4, and 5 by LCO 3.3.1, Reactor Trip System Instrumentation.

(continued)

BASES (continued)

ACTIONS

-----------------------NOTE-----------------------
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

A.1

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. With no audible indication in the containment or control room, prompt notification of a potential problem cannot be assumed. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended within 15 minutes.

B.1 and B.2

With no source range neutron flux monitors OPERABLE, there is no direct means of detecting changes in core reactivity and CORE ALTERATIONS must be suspended immediately. Actions must also be initiated within 15 minutes to restore one source range neutron monitor to OPERABLE status. The means to ensure that the reactor remains subcritical is to perform SR 3.9.1.1. Performing SR 3.9.1.1 verifies that the required boron concentration exists. The Completion Time of 4 hours is based on the practical amount of time necessary to perform the surveillance. Performing the surveillance once per 12 hours thereafter ensures that unpredictable changes in boron concentration are not occurring.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

Performing a CHANNEL CHECK provides assurance that the channels have not drifted outside their limits. A CHANNEL CHECK is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two channels of indication should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the electronics have drifted outside of their limits. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their trip setpoints. The frequency of 12 hours is based on engineering judgment, and is consistent with LCO 3.3.2, Engineered Safety Features Actuation System Instrumentation.

SR 3.9.3.2

Verification of audible indication in the containment and control room assures that the operators and personnel in containment can be aware of any changes in core reactivity. The frequency of 12 hours is a reasonable time to detect any abnormalities in the system.

SR 3.9.3.3

The performance of an ANALOG CHANNEL OPERATIONAL TEST provides assurance that the analog process control equipment and trip setpoints are within limits. This test is a periodic check of the process control equipment. When the channel is placed in the test condition, the input from the transmitter is removed and the trip output is isolated. This allows a test signal to be introduced into the instrument loop. The input can be measured, thus noting the accuracy of the signal conditioning of the process control modules upstream. The trip setpoint of the channel can be determined by varying the input and observing the trip status. The frequency of 7 days is based on engineering judgment and ensures systematic verification of the channels.

REFERENCES

1. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Building Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel within containment, a release of fission product radioactivity within containment will be restricted from leakage to the environment when the LCO requirements are met. The containment structure serves to contain fission product radioactivity which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, this structure provides radiation shielding from the fission products which 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. The equipment hatch is normally closed. Plant drawings and procedures dictates how many bolts are required for closure and their orientation.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during unit operation. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment closure is required.

During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an airlock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel within containment, the door interlock mechanism may remain disabled, allowing one airlock door to remain open for extended periods, providing that the second door is maintained closed and/or access is administratively controlled by a dedicated individual stationed at the air lock.

(continued)

BASES (continued)

BACKGROUND (continued)

The requirements on containment building penetration closure ensure that a release of fission product radioactivity within containment will be restricted from leaking 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 two purge and exhaust penetrations must be isolated or isolable and the other penetrations which provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by OPERABLE automatic containment vent isolation valves or by an isolation valve, blind flange, or equivalent. Equivalent isolation methods may include a closed system within containment or a material which can provide a temporary, atmospheric pressure, ventilation barrier for the reactor building penetrations during fuel movements. An example of such a material which has been approved for this use is a silicone foam sealant (Ref. 2).

There are no manual valves to isolate the purge and exhaust penetrations. Conversely, all other penetrations which provide direct access, such as drain, vent, or test lines, have no automatic isolation valves.

OPERABILITY of the Containment Vent Isolation System ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radioactivity levels within containment. OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

Redundant and independent gaseous and particulate radioactivity monitors measure the radioactivity levels of the containment atmosphere, each of which will initiate both trains of automatic Containment Vent Isolation upon detection of high particulate or gaseous radioactivity. Redundant and independent gaseous radioactivity monitors measure the radioactivity levels of the containment purge exhaust, each of which will initiate its associated train of automatic Containment Vent Isolation upon detection of high gaseous radioactivity. Automatic Containment Vent Isolation consists of two redundant and independent trains of logic circuitry and actuation relays. Both trains of actuation relays in the Containment Vent Isolation System are also actuated by main control room hand switches. A loss of power to any of these monitors will result in a containment vent isolation.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel within containment, the most severe radiological consequences results from a fuel handling accident. The fuel handling accident is a Condition IV postulated event which involves damage to irradiated fuel (Ref. 3). Fuel handling accidents, analyzed in FSAR Section [15.5.6], include dropping a single fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies (Ref. 4). The requirements of this LCO, LCO 3.9.7 and LCO 3.9.9 ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, does not result in doses in excess of the guideline values specified in 10 CFR 100 and Standard Review Plan Section 15.7.4, Rev. 1, (Ref. 3). This specification contains systems and components that are part of the primary success path which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, the requirements for containment penetrations satisfy the requirements of Criterion 3 of the NRC Interim Policy Statement (Ref. 5).
LCOs	This LCO minimizes the consequences of a fuel handling accident in containment by ensuring that the release of fission product radioactivity is confined to containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the containment purge and exhaust penetrations. For the containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Vent Isolation System.

(continued)

BASES (continued)

LCOs (continued)

Each required train of the Containment Vent Isolation System automatically isolate each containment vent and purge penetrations upon receipt of a manual initiation signal or a high radiation signal to prevent the release of fission product radioactivity to the environment in excess of the 10 CFR 100 limits.

The radiation monitoring equipment required for train A of the Containment Vent Isolation System to be OPERABLE includes the particulate and gaseous radiation monitoring channels of either containment atmospheric radiation monitor and the train A purge exhaust gaseous radiation monitor. The radiation monitoring equipment required for train B of the Containment Vent Isolation System to be OPERABLE includes the particulate and gaseous radiation monitoring channels of either containment atmospheric radiation monitor and the train B purge exhaust gaseous radiation monitor.

Loss of OPERABILITY of particulate or gaseous radiation monitoring channels in one of the containment atmospheric radiation monitors or loss of OPERABILITY of either purge exhaust gaseous monitor results in Condition A. Loss of OPERABILITY of particulate or gaseous radiation monitoring channels in Both of the containment atmospheric radiation monitors or both of the purge exhaust gaseous radiation monitors results in Condition B.

APPLICABILITY

The containment building penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel within containment since this is when there is a potential for a fuel handling accident. Due to the RCS pressure and temperature limitations in MODE 6, the requirements of LCO 3.6.1, Containment, have been relaxed because the containment cannot be pressurized from an accident condition in MODE 6.

(continued)

BASES (continued)

ACTIONS

A.1

Loss of one train of the Containment Vent Isolation System results in reduced redundancy in the ability to mitigate the results of a fuel handling accident in containment. Therefore, the affected train must be returned to an OPERABLE status in a short time. A Completion Time of 4 hours is chosen to be consistent with LCO 3.6.6, Containment Isolation Valves, which allows 4 hours to restore the loss of one of two redundant automatic containment isolation valves.

B.1

To continue operations involving CORE ALTERATIONS and positive reactivity additions when the equipment hatch, airlocks or any containment penetration providing direct access from the containment atmosphere to the outside atmosphere is not in its required status the affected penetration must be restored to its required status within 15 minutes. The specified Completion Time is consistent with the time required to implement the required action. This condition results in a loss of the ability to prevent or mitigate a release of fission product radioactivity to the environment and the required action is sufficient to restore ability to prevent or mitigate releases.

To continue operations involving CORE ALTERATIONS and positive reactivity additions when both of the required Containment Vent Isolation System trains are inoperable or when the action required for the loss of one train of the Containment Vent Isolation System cannot be implemented within its specified Completion Time, all Containment Vent Isolation Valves must be closed and be maintained closed until both required trains of the Containment Vent Isolation System are restored to an OPERABLE condition. The Completion Time of 15 minutes for this action is consistent with the time required to implement the required action. This condition results in a loss or unacceptably prolonged reduction of the ability to prevent or mitigate a release of fission product radioactivity to the environment and the required action is sufficient to restore ability to prevent or mitigate releases.

(continued)

BASES (continued)

ACTIONS (continued)

B.2

Suspending CORE ALTERATIONS and positive reactivity additions as soon as safely possible is a suitable alternative to the actions specified in B.1 for any of the applicable conditions since this action will reduce the possibility of a fuel handling accident while the containment is not closed or capable of being closed. The required Completion Time of 15 minutes is considered to be the time needed to safely implement this action.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

Performing a CHANNEL CHECK provides assurance that the channels have not drifted outside their limits. A CHANNEL CHECK is the comparison of the indicated parameter values for each of the functions. It is based on the assumption that the two channels of indication should be reading approximately the same. Agreement is based on a combination of the channel instrument uncertainties, including control isolation, indication, and readability. If a channel is outside of the match criteria, it may be an indication that the transmitter or the electronics have drifted outside of their limits. If the channels are within the match criteria, it is a reasonable assumption that the channels are within specification with respect to their trip setpoints. The frequency of 12 hours is based on engineering judgment, and is consistent with LCO 3.3.2, Engineered Safety Features Actuation System Instrumentation.

SR 3.9.4.2

The surveillance verifies that each of the containment building penetrations, required to be in its closed position, is in that position. As such, this surveillance ensures that a postulated fuel handling accident which involves a release of fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

The surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment. The surveillance interval is based on engineering judgment, industry operating experience, and system reliability.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.3

SR 3.9.4.3 verifies that containment vent isolation occurs on manual initiation or upon the introduction of a high radiation test signal from each purge and containment atmosphere radiation monitoring channel. This surveillance verifies, in addition to the electronics, the actual closing of the isolation valves. The surveillance is performed every 14 days during CORE ALTERATIONS or movement of irradiated fuel in the containment. The surveillance interval is based on engineering judgment, industry operating experience, and system reliability.

SR 3.9.4.4

The performance of an ANALOG CHANNEL OPERATIONAL TEST provides assurance that the analog process control equipment and trip setpoints are within limits. The frequency of 31 days is based on engineering judgment and is consistent with the LCO 3.3.2, Engineered Safety Features Actuation System Instrumentation.

SR 3.9.4.5

A CHANNEL CALIBRATION provides assurance of instrumentation channel OPERABILITY by performing a complete check of the process control instrument loop and the sensor using a NBS traceable radioactivity source. Completion of the CHANNEL CALIBRATION results in the channel being properly adjusted and expected to remain within the allowable value until the next scheduled surveillance. As such, the frequency of 18 months is based on engineering judgment, and has been shown to be acceptable through industry operating experience.

(continued)

BASES (continued)

REFERENCES

1. Title 10 Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation."

2. "Use of Silicone Sealant to Maintain Containment Integrity - ITS," GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

3. NUREG-0800, Standard Review Plan Section 15.7.4, Radiological Consequences of Fuel Handling Accidents, Rev. 1, July 1981

4. Watts Bar FSAR Section [15.5.6], Environmental Consequences of a Posulated Fuel Handling Accident.

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal and Coolant Circulation - High Water Level

BASES

BACKGROUND

The main purpose of the Residual Heat Removal (RHR) System is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) when RCS pressure and temperature are below approximately [380] psig and [350]°F, respectively (Ref. 1). Heat is transferred from the RCS by circulating reactor coolant through the RHR System where the heat is transferred to the Component Cooling Water System via the RHR heat exchangers.

In the decay heat removal mode of operation, the RHR System takes suction from loop 4 RCS hot legs. Flow from the RHR pumps is discharged through heat exchangers and/or bypasses, and is returned to the RCS cold legs. This arrangement provides two redundant RHR loops except for the suction line. Operation of the RHR System for normal cooldown/decay heat removal is manually accomplished from the control room.

APPLICABLE SAFETY ANALYSES

With the unit in MODE 6, the RHR System is not required to mitigate any events or accidents evaluated in the safety analyses (Ref. 2). However, the importance of the RHR System has been addressed by the NRC. The NRC Interim Policy Statement requires that the RHR System in MODE 6 be retained in the Technical Specifications, even though none of the selection criteria were satisfied (Ref. 3).

LCOs

Only one RHR loop is required for decay heat removal in MODE 6 with water level greater than or equal to 23 feet above the top of the reactor vessel flange. At least one RHR loop must be OPERABLE and in operation in order to:

a. Provide forced circulation for decay heat removal,

b. Provide mixing of borated coolant to minimize the possibility of criticality, and

c. Provide indication of average reactor coolant temperature.

(continued)

BASES (continued)

LCOs (continued)

The requirements of this LCO are derived primarily from experience with decay heat removal in shutdown modes of operation. The principal purpose of this Specification is to assure the capability to remove decay heat and to control RCS flow, temperature, and chemistry.

To allow maximum unit flexibility, the required operating RHR loop may be removed from service for up to 1 hour per 2 hour period. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and 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.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6 with the water level greater than or equal to [23] feet above the top of the reactor vessel flange to provide decay heat removal and forced circulation in the RCS. This water level provides the required radiation shielding during fuel handling operations, a relatively large heat sink and iodine gas filtration for the postulated fuel handling accident. An uncontrolled boron dilution transient is prevented from occurring in MODE 6 (Ref. 2). This event is protected against by controls which isolate all sources of unborated water from the RCS. Requirements for the RHR System in other MODES are covered by LCOs in Chapter 3.4, Reactor Coolant System, and in Chapter 3.5, Emergency Core Cooling System.

(continued)

BASES (continued)

ACTIONS

-------------------------NOTE---------------------------
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

A.1

With no RHR loop in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Although sources of unborated water have been locked out, reduced boron concentrations can be achieved by the addition of water with lower boron concentration than that contained in the RCS. Therefore, actions which reduce boron concentration shall be suspended immediately. The 15 minute Completion Time is a reasonable time to take these actions.

A.2

With no RHR loop OPERABLE or in operation, actions shall be taken within 15 minutes to suspend operations involving an increase in reactor decay heat load. 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. Suspending any operation which would increase decay heat load, such as loading a fuel assembly, is a prudent engineering judgment under this condition. A minimum refueling water level of [23] feet above the reactor vessel flange provides an adequate available heat sink.

A.3

With no RHR loop OPERABLE or in operation, actions shall be taken and continued without interruption to restore an RHR loop to operation and OPERABLE status. With the unit in MODE 6, 15 minutes is a reasonable time to initiate corrective actions. The corrective actions should be continued until the loop is restored to OPERABLE status.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This surveillance verifies that the RHR loop is operating and circulating reactor coolant to ensure the capability of the RHR System to maintain compliance with unit design limits. The required RHR loop reactor coolant flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability to maintain the reactor coolant temperature rise through the core within design limits. Flow may be injected through only 2 RCS cold leg flow paths as necessary to support other activities. The frequency of 12 hours is based on engineering judgment and ensures that RHR loop operation and flow is checked at adequate intervals.

REFERENCES

1. Watts Bar FSAR, Section [5.5.7]
2. Watts Bar FSAR, Section [15.5.6]
3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The main purpose of the Residual Heat Removal (RHR) System is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) when RCS pressure and temperature are below approximately [380] psig and [350]°F, respectively (Ref. 1). Heat is transferred from the RCS by circulating reactor coolant through the RHR System where the heat is transferred to the [Component Cooling Water System] via the RHR heat exchangers.

In the decay heat removal mode of operation, the RHR System takes suction from one of the RCS hot legs. Flow from the RHR pumps is discharged through heat exchangers or bypasses, and is returned to the RCS cold legs. This arrangement provides two redundant RHR loops with the exception of the suction line. Operation of the RHR System for normal cooldown/decay heat removal is manually accomplished from the control room.

APPLICABLE SAFETY ANALYSES

With the unit in MODE 6, the RHR System is not required to mitigate any events or accidents evaluated in the safety analyses (Ref. 2). However, the importance of the RHR System has been addressed by the NRC. The NRC Interim Policy Statement requires that the RHR System in MODE 6 be retained in the Technical Specifications, even though none of the selection criteria were satisfied (Ref. 3).

LCOs

Only one RHR loop is required for decay heat removal in MODE 6 with water level less than 23 feet above the top of the reactor vessel flange. To increase reliability, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to:

a. Provide forced circulation for decay heat removal,

b. Provide mixing of borated coolant to minimize the possibility of criticality, and

c. Provide indication of average reactor coolant temperature.

(continued)

BASES (continued)

LCOs (continued)

The requirements of this LCO are derived primarily from experience with decay heat removal in shutdown modes of operation. The principal purpose of this specification is to assure the capability to remove decay heat and to control RCS flow, temperature, and chemistry with low water level.

APPLICABILITY

Two RHR loops are required to be OPERABLE and one RHR loop must be in operation in MODE 6 with the water level less than 23 feet above the top of the reactor vessel flange to provide decay heat removal and forced circulation in the RCS. With water level less than 23 feet above the top of the reactor vessel flange, fuel movements are precluded by LCO 3.9.7. An uncontrolled boron dilution transient is prevented from occurring in MODE 6 (Ref. 2). This event is protected against by controls which isolate all sources of unborated water from the RCS. Requirements for the RHR System in other MODES are covered by LCOs in Chapter 3.4, Reactor Coolant System, and in Chapter 3.5, Emergency Core Cooling System.

ACTIONS

A.1

With one RHR loop inoperable, actions shall be taken to restore the RHR loop to OPERABLE status. With the unit in MODE 6, corrective actions must be initiated within 15 minutes. The 15 minutes is a reasonable time to initiate corrective actions. The corrective actions should be continued until the inoperable loop is restored to OPERABLE status.

A.2

Alternate decay heat removal capabilities shall be established to provide another method of decay heat removal in the event one of the OPERABLE RHR loops becomes inoperable. This capability may be established by aligning other pumps and systems, such as charging or safety injection pumps and the CVCS, to provide reactor coolant circulation or by raising the water level to a minimum of [23] feet above the reactor vessel flange. If the chosen alternate means to remove decay heat is to raise the water level to a minimum of [23] feet above the reactor vessel flange, the Applicability will change to that of LCO 3.9.5

(continued)

BASES (continued)

ACTIONS (continued)

where only one RHR loop is required to be OPERABLE and in operation. A Completion Time of 7 days is, based on experience, consistent with the time necessary to disassemble, repair and return an RHR pump to service. After 7 days, the alternate decay heat removal capabilities shall be available for use.

-------------------------NOTE--------------------------
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

B.1

With no RHR loop in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Although sources of unborated water have been locked out, reduced boron concentrations can be achieved by the addition of water with lower boron concentration than that contained in the RCS. The Completion Time of 15 minutes is sufficient to suspend actions which would reduce boron concentration.

B.2

With no RHR loop in operation or with both RHR loops inoperable, actions shall be taken and continued without interruption to restore one RHR loop to OPERABLE status and operation. These actions should be initiated immediately. As the unit is in Conditions A and B simultaneously, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished as quickly as possible.

B.3

With no RHR loop in operation or both RHR loops inoperable, actions shall be initiated within 15 minutes to implement alternate decay heat removal and actions should continue to affect the removal of decay heat from the reactor coolant as quickly as possible. Decay heat removal may be accomplished by use of the charging or safety injection pumps through the CVCS with consideration for the boron concentration or by raising the water level to a minimum of [23] feet above the

(continued)

BASES (continued)

ACTIONS (continued)

reactor vessel flange. Raising the water level to a minimum of 23 feet above the reactor vessel flange would provide an adequate heat sink for decay heat removal. Raising the water to this level also will change the Applicability to that of LCO 3.9.5 where only one RHR loop is required to be OPERABLE and in operation. The method used to remove decay heat should be the most prudent and safe choice based upon unit configuration. The choice could be different if the reactor vessel head is in place than if the reactor vessel head is removed.

With the unit in MODE 6, 15 minutes is an adequate time to initiate action to implement alternate decay heat removal. The established method should be placed in operation as quickly as possible to affect the removal of decay heat.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This surveillance verifies that the RHR loop is operating and circulating reactor coolant to ensure the capability of the RHR System to maintain compliance with unit design limits. The required RHR loop reactor coolant flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability to maintain the reactor coolant temperature rise through the core within design limits, and to prevent thermal and boron stratification in the core. Flow may be provided to the core through only 2 cold leg injection paths to support other activities as necessary.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR loop flow rate determination must also consider the RHR pump suction requirements. At this water level, if the flow rate demand is too high, the RHR System can experience vortexing which could result in pump cavitation. Care must be taken in determining that the RHR loop flow rate, when operating with water level in this region, is adequate to prevent loss of the RHR pump and subsequent loss of the RHR loop for decay heat removal. The frequency of 12 hours is based on engineering judgment and ensures that flow is checked and temperature monitored at adequate intervals.

(continued)

BASES (continued)

SURVEILLANCE REQUIRMENTS (continued)

SR 3.9.6.2

Verifying that the second RHR loop is OPERABLE ensures that the single failure criterion is met. The requirement also ensures that the second OPERABLE RHR loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. The surveillance of 7 days interval ensures that the required flow path can be made available and is based on engineering judgment.

REFERENCES

1. Watts Bar FSAR, Section [5.5.7]

2. Watts Bar FSAR, Section [15.5.6]

3. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

Requirements on water level in the containment, the refueling cavity, the refueling canal, the fuel transfer canal, and the spent fuel pool during refueling ensure that sufficient water depth is available to remove 99% of the iodine activity from the fuel pellet/cladding gap. The iodine activity is assumed to be released during the postulated rupture of an irradiated fuel assembly in containment. The fuel pellet/cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

Although not included in this LCO, the water level requirements also ensure that, during movement of irradiated fuel assemblies from the reactor vessel to the refueling canal transfer device, the maximum dose rate of 2.5 mr/hr at the water surface is not exceeded. A water level of greater than [23] feet above irradiated fuel provides adequate radiation shielding. The minimum water level above the irradiated fuel assemblies is consistent with the assumptions of the accident analysis (Ref. 3).

For movement of fuel assemblies or RCCAs, a minimum water level of [23] feet above the top of the reactor vessel flange is required. This level ensures a minimum of [10] feet of water over the active portions of the fuel assemblies during withdrawal and transfer activities.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies and RCCAs, the water level in the refueling cavity and refueling canal is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by NRC Regulatory Guide 1.25 (Ref. 1). A minimum water level of [23] feet (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released, from the gap of all the dropped fuel assembly rods, is retained by the refueling cavity water.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The fuel handling analysis inside containment (Ref. 3) assumes: 1) the accident occurs [100] hours after reactor shutdown, 2) the minimum water depth above the top of the ruptured fuel rod is [23] feet, 3) fission product inventories match those for full power operation at end of core life, and 4) containment isolation and exhaust systems are properly operating. With a minimum water level of [23] feet, 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. 4).

The minimum water level is a process variable that is an initial condition of a design basis accident or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, the minimum water levels satisfy the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 5).

APPLICABILITY

LCO 3.9.7, Refueling Cavity Water Level, is applicable when moving fuel assemblies or RCCAs within containment with irradiated fuel assemblies in containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident.

LCOs

The requirement for a minimum of [23] feet of water above the reactor vessel flange provides adequate water coverage for the postulated fuel handling accident analysis.

(continued)

BASES (continued)

ACTIONS

---NOTE---
Performance of Required Actions to suspend CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe conservative condition.

A.1

With a water level of less than [23] feet above the top of the reactor vessel flange, all operations involving movement of fuel assemblies and RCCAs shall be suspended immediately. The suspension of fuel and RCCA movement shall not preclude completion of movement to a safe conservative position where the fuel assembly or RCCA cannot be dropped or damaged, causing a fuel handling accident.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of [23] feet above the top of the reactor vessel flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. With water at the required level above the top of the reactor vessel flange, mitigation is provided for a postulated fuel handling accident inside containment which results in damaged fuel rods (Ref. 3).

The frequency of 24 hours ensures that the water is at the required level prior to entering the LCO Applicability and monitors the level to detect any unplanned changes in water level. The frequency is considered, by engineering judgment, to be adequate to determine and monitor the water level.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," United States Nuclear Regulatory Commission, March 23, 1972.

2. Title 10 Code of Federal Regulations, Part 20, Section 20.101(a), "Radiation Dose Standards for Individuals in Restricted Areas."

3. Watts Bar FSAR, Section [15.5.6]

4. [Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-7828, Radiological Consequences of a Fuel Handling Accident, December, 1971.]

5. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

B 3.8 REFUELING OPERATIONS

B 3.8.8 Decay Time

BASES

BACKGROUND

The primary purpose of the decay time requirement is to ensure that the fission product inventories assumed in the fuel handling accident analysis are met. As soon as the reactor is subcritical, the quantity of fission products in the core decreases as the fission products undergo natural radioactive decay. As long as the reactor remains subcritical, this decrease will continue and the radiation levels will also decrease. In addition, to minimize personnel radiation exposures during refueling operations, it is necessary to allow the fission products to decay for an appropriate time.

APPLICABLE SAFETY ANALYSES

The fuel handling accident is the postulated event of concern in MODE 6 during fuel handling operations (Ref. 1). It establishes the minimum decay time. A fuel assembly is assumed to fall onto the refueling cavity/canal floor and rupture all the fuel rods of that fuel assembly. The dropped fuel assembly is assumed to be the assembly with the highest fission product inventory. The fission product inventories are those assumed to be present [100] hours after the reactor becomes subcritical.

Decay time is a process variable that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As such, the decay time satisfies the requirements of Criterion 2 of the NRC Interim Policy Statement (Ref. 2).

LCOs

The LCO requires that the reactor be subcritical for at least [100] hours prior to commencing CORE ALTERATIONS. The requirement to be subcritical for greater than or equal to [100] hours ensures that the fission product radioactivity has undergone natural radioactive decay and that the consequences of a fuel handling accident will be within the bounds of the safety analysis.

(continued)

BASES (continued)

APPLICABILITY	This LCO is only applicable during CORE ALTERATIONS. These operations are not performed in any other operational MODE.
ACTIONS	A.1 With the reactor subcritical for less than [100] hours, there shall be no operations involving CORE ALTERATIONS. This will preclude a fuel handling accident with fuel containing more fission product radioactivity than assumed in the safety analysis.
SURVEILLANCE REQUIREMENTS	SR 3.9.8.1 Once each shutdown prior to the initial start of CORE ALTERATIONS, the reactor must be determined to be subcritical for greater than or equal to [100] hours by verifying the date and time that the reactor achieved subcritical conditions.
REFERENCES	1. Watts Bar FSAR, Section [15.5.6], Environmental Consequences of a Postulated Fuel Handling Accident. 2. 52FR3788, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," US Nuclear Regulatory Commission, February 6, 1987.

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT
RELOCATED TECHNICAL SPECIFICATIONS

ENCLOSURE 3

LCOs TO BE RELOCATED OUTSIDE THE WBN TS

WBN 1985 Draft #	
3.1.2.1	Boration Systems Flow Paths - Shutdown
3.1.2.2	Boration Systems Flow Paths - Operating
3.1.2.3	Charging Pumps - Shutdown
3.1.2.4	Charging Pumps - Operating
3.1.2.5	Borated Water Sources - Shutdown
3.1.2.6	Borated Water Sources - Operating
3.1.3.3	Position Indication Systems - Shutdown
3.1.3.4	Rod Drop Time
3.3.3.2	Moveable Incore Detectors
3.3.3.3	Seismic Monitoring
3.3.3.4	Meteorological Instrumentation
3.3.3.7	Fire Detection Instrumentation
3.3.3.8	Radioactive Liquid Effluent Monitoring Inst.
3.3.3.9	Radioactive Gaseous Effluent Monitoring Inst.
3.3.3.10	Loose Parts Detection
3.3.4	Turbine Overspeed Protection
3.4.2.1	Safety Valves, Shutdown
3.4.5	Steam Generator
3.4.7	RCS Chemistry
3.4.9.2	PRZR Temp Limits
3.4.10	RCS Structural Integrity
3.4.11	RCS Vent Paths
3.5.12	UHI - System removed at WBN
3.6.1.2	Containment Leakage - The LCO will be relocated but Pa, La, L_t, and L_D will be retained in 4.0 Design Features
3.6.1.6	Containment Vessel Structural Integrity
3.6.1.7	Containment Shield Building Structural Integrity
3.6.3	Containment Isolation Valves - relocate response times
3.6.5.4	Inlet Door Position Monitoring

LCOs TO BE RELOCATED OUTSIDE THE WBN TS

	WBN 1985 Draft #	
	3.7.2	Steam Generator Pressure/Temperature Limitations
	3.7.6	Flood Protection
	3.7.9	Snubbers
	3.7.10	Sealed Source Contamination
	3.7.11.1	Fire Suppression Water System
	3.7.11.2	Spray and/or Sprinkler System
	3.7.11.3	CO_2 Systems
	3.7.11.4	Fire Hose Stations
	3.7.1.2	Fire-Rated Assemblies
	3.7.1.3	Area Temperature Monitoring
	3.8.4.1	Containment Penetration Conductor Overcurrent Protection Devices
	3.8.4.2	Motor-Operated Valve Thermal Overload Bypass Devices
	3.9.5	Communications
	3.9.6	Refueling Machine
	3.9.7	Crane Travel - Spent Fuel Storage Pool Building
	3.10.1	Shutdown Margin - Not needed - due to being covered under 3.1.1.1 or 3.1.5
*	3.11.1.1	Liquid Effluent Concentration
*	3.11.1.2	Dose-Liquid Effluents
*	3.11.1.3	Liquid Radwaste Treatment System
*	3.11.1.4	Liquid Holdup Tanks
*	3.11.2.1	Gaseous Effluent - Dose Rate
*	3.11.2.2	Dose-Noble Gases
*	3.11.2.3	Dose-Idodine-131 and 133, Tritium, and Radioactive Material in Particulate Form
*	3.11.2.4	Gaseous Radwaste Treatment System
*	3.11.2.5	Explosive Gas Mixture
*	3.11.2.6	Gas Decay Tanks
*	3.11.3	Solid Radioactive Wastes
*	3.11.4	Total Dose
*	3.12.1	Monitoring Program
*	3.12.2	Land Use Census
*	3.12.3	Interlaboratory Comparison Program

* These LCO's will be relocated and will be controlled as a program requirement of 5.9, Programs, of the WBN TS.