LAR 93-14R1

RIVER BEND

CHAPTER 1

ATTACHMENT 1: NOT USED
ATTACHMENT 2: ITS - PSTS COMPARISON DOCUMENT

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ATTACHMENT 2

ITS - PSTS

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REVISION 1

CHAPTER 1 REVISED PAGES

2A: MARKUP OF ITS

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MARKUP OF ITS

1.1 Definitions

CHANNEL CALIBRATION (continued)

remaining adjustable devices in the channel. ~~Whenever a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element.~~ The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

C7

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

~~a. Analog channels~~—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips.

C8

~~b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions.~~

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or ~~other components affecting reactivity~~ within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range

C9

3.1 #15

The following exceptions are not considered to be CORE ALTERATIONS;
a.

(continued)

1.1 Definitions

CORE ALTERATION (continued)

monitors, traversing incore probes, or special movable detectors (including undervessel replacement)~~, is not considered a CORE ALTERATION. In addition,~~ b. control rod movement with other than the normal control rod drive ~~is not considered a~~ CORE ALTERATION provided there are no fuel assemblies in the associated core cell.

3.1 #15

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification ~~5.9.1.6.~~ 5.6.5. Plant operation within these limits is addressed in individual Specifications.

1.0 #15 P3

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." ~~or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977]~~.

1.0 #11 B1

~~[$\bar{E}$—AVERAGE DISINTEGRATION ENERGY~~

~~$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.]~~

B1

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their

(continued)

1.1 Definitions

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME (continued)

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 means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME

P2

B1

The EOC-RPT SYSTEM RESPONSE movement of TIME shall be that time interval from initial ~~signal generation by~~ ~~[~~the associated turbine stop valve ~~limit switch~~ or ~~from when~~ the turbine control valve ~~hydraulic oil control oil pressure drops below the pressure switch setpoint]~~ to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured~~,~~. ~~[except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value].~~

B1

ISOLATION SYSTEM RESPONSE TIME

C1

The ISOLATION SYSTEM initiation RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation setpoint at the channel sensor until the isolation valves travel to their required positions. ~~Times shall include diesel generator starting and sequence loading delays, where applicable.~~ The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

1.0 #14

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or

1.0 #12

La

The maximum allowable primary containment leakage rate, La, shall be 0.26% of primary containment air weight per day at the calculated peak containment pressure (Pa).

(continued)

1.1 Definitions

LEAKAGE (continued)	2. LEAKAGE into the drywell 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;
	b. Unidentified LEAKAGE All LEAKAGE into the drywell that is not identified LEAKAGE;
	c. Total LEAKAGE Sum of the identified and unidentified LEAKAGE;
	d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
[LINEAR HEAT GENERATION RATE (LHGR) B1	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.]
LOGIC SYSTEM FUNCTIONAL TEST C2	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated ~~device~~ device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
C10 ~~equipment~~ 1.0 #9 -- device	
[MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) B1	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.]

(continued)

1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR) B1	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core ~~[for each class of fuel]~~. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY C3 C5 division	A system, subsystem, [redacted], component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, [redacted], component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
~~PHYSICS TEST~~ P1	~~PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:~~ ~~a. Described in Chapter [14, Initial Test Program] of the FSAR;~~ ~~b. Authorized under the provisions of 10 CFR 50.59; or~~ ~~c. Otherwise approved by the Nuclear Regulatory Commission.~~
~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR 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~~ 1.0 #13 5.0 #127

(continued)

1.0 #13

5.0 #127

1.1 Definitions

5.6.6. P3 LAR 93-14 R1

~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)~~	~~shall be determined for each fluence period in accordance with Specification 5.9.1.7. Plant operation within these operating limits is addressed in LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."~~
RATED THERMAL POWER (RTP)	B1 RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[3833]~~ 2894 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. 1.0 #10 C4 With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

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CHAPTER 2

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ATTACHMENT 2

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CHAPTER 2 REVISED PAGES

2A: MARKUP OF ITS

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MARKUP OF ITS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

B1 MCPR shall be ≥ [1.07] for two loop recirculation operation or ≥ [1.08] for single loop recirculation operation. C6

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

C7

Reactor steam dome pressure shall be ~~maintained~~ ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

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

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.0 #7a

P6

2.2.3 Within 24 hours, notify the ~~[General]~~ Manager ~~- Nuclear~~ Plant ~~and Vice President - Nuclear Operations]~~ ~~and the~~ ~~[offsite reviewers specified in~~ ~~Specification 5.5.2, "[Offsite] Review and Audit"]~~.

5.0 #73

P6 and the corporate executive responsible for overall plant nuclear safety.

E1 ~~Nuclear Review Board (NRB)~~

(continued)

2.0 SLs

2.2 SL Violations (continued)

NRB ~~(B)~~

2.0 #7a

Pb

5.0 #73

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the ~~[offsite reviewers specified in Specification 5.5.2], and the [General]~~ Manager ~~- Nuclear~~ Plant, and ~~Vice President - Nuclear Operations]~~. the corporate executive responsible for overall plant nuclear safety.

2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.

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BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

C9

2.0 #76

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR SL is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other ~~all the~~ LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR SL.

C9

P4

2.1.1.1 Fuel Cladding Integrity ~~(General Electric Corporation (GE) Fuel)~~ P5

2.0 #76

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig ~~or~~ and core flows ≥ 10% of rated flow. For operation at low pressures or ~~and~~ low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

2.0 #76

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel~~, thus maintaining a coolable geometry~~ in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

C7

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES. ~~However, in MODES 3, 4, and 5, with the reactor shut down, it is unlikely that fuel cladding integrity SLs would be violated.~~

C12

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~4~~ 3).

P4

2.2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4 ~~5~~). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

P4

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If any SL is violated, the ~~appropriate senior management of the nuclear plant and the utility~~ General Manager Plant Operations and the Vice President - Operations shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.0 #7a

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC~~, the senior management of the nuclear plant, and the utility Vice President, Nuclear Operations~~. This requirement is in accordance with 10 CFR 50.73 (Ref. ~~6~~ 5). A copy of the report shall also be submitted to the

C2 P4

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR II," (latest approved revision). P7
3. ~~XN-NF524(A), Revision 1, November 1983.~~
3 ~~4~~. 10 CFR 50.72.
4 ~~5~~. 10 CFR 100.
5 ~~6~~. 10 CFR 50.73.

P4

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

C1 The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs). C3 ~~During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability.~~

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core ~~is~~ may be done under C5 LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." C1 Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the C13 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, "Reactor Site Criteria" (Ref. 4).

2.0 #7c

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial ~~action~~ actions and also ensures that the probability of an accident occurring during this period is minimal.

C14

2.2.3

If any SL is violated, the ~~appropriate senior management of the nuclear plant and the utility~~ General Manager Plant Operations and the Vice President - Operations shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.0 #7a

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC~~, the senior management of the nuclear plant, and the utility Vice President—Nuclear Operations~~. ~~This requirement is in accordance with 10 CFR 50.73 (Ref. 8).~~ A copy of the report shall also be submitted to the General Manager Plant Operations and the Vice President - Operations

C2

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III. ~~Article NB-7000.~~

P9

(continued)

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SECTION 3.0

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
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ATTACHMENT 1

ITS - PSTS

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1A: MARKUP OF CTS

1B: NOT USED

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LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

3.0 #1

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

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INSERT 2D

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0 #1

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

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MARKUP OF ITS

3.0 LCO APPLICABILITY

3.0 #1

C2

LCO 3.0.4 (continued) C8 specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited 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 solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

5.0 #104 5.5.10 P1

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification ~~5.6~~, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

(continued)

3.0 SR APPLICABILITY

SR 3.0.3 (continued)

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. ~~The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.~~

C3

SR 3.0.4

Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent ~~passage through or to~~ entry into MODES or other specified conditions in ~~compliance~~ the Applicability that are required to comply with ~~Required Actions~~ ACTIONS or that are part of a shutdown of the unit.

C4

C8

3.0 #1

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

BASES

LCO 3.0.3 (continued)

assemblies in the associated fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.~~7~~ 6 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.~~7~~ 6 of "Suspend movement of irradiated fuel assemblies in the associated fuel storage pool(s)" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a ~~different~~ MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before ~~unit startup~~ entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES

(continued)

BASES

#/ C2

LCO 3.0.4 (continued) C8

or other specified conditions in the Applicability that result from ~~a normal~~ any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

3.0 #1

INSERT B6A

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment

(continued)

LAR 93-14R1

INSERT B6A

3.0 #1

LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. ~~[In some cases (e.g., ...) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]~~

B1

BASES

LCO 3.0.6 (continued) C2

declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

5.0 #104

5.5.10 P1

Specification ~~5.8~~ 5.5.10, "Safety Function Determination Program" (SFDP), ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will

(continued)

BASES

SR 3.0.3 (continued)

Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. ~~This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.~~

C13
INSERT B14 A

C8

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. INSERT B14B

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

3.0 #1
INSERT B14C

LAR 93-14R1

, which states that

3.0 #3

INSERT B14A

However, in certain circumstances failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1. Surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions in the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

3.0 #1

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

INSERT B14B

In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

INSERT B14C

3.0 #1

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

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RIVER BEND

SECTION 3.1

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.1 REVISED PAGES

1A: MARKUP OF CTS

1B: NOT USED

1C: NOT USED

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

SURVEILLANCE REQUIREMENTS (Continued) LA4 or can be aligned to the correct position.

SR 3.1.7.5 2. Determining*, that the available weight of Boron-10 is greater than or equal to 143 lbs, the percent weight concentration of sodium pentaborate in solution is equal to or less than 9.5% by weight, and the minimum required solution volume.

SR 3.1.7.6 3. Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position,

SR 3.1.7.7 4. Determining that the Standby Liquid Control System satisfies the following equation:

$(C)(E) \geq 413$

Where:

C = sodium pentaborate concentration, in weight percent, as determined per specification 4.1.5.b.2.

E = Boron-10 enrichment, in atom percent**.

c. SR 3.1.7.7 Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1220 psig is met.

d. At least once per 18 months during shutdown by; LA2

SR 3.1.7.8 1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. LA1 The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. LA3 Both injection loops shall be tested in 36 months.

SR 3.1.7.5 *This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 45°F. A1

SR 3.1.7.9 **The Boron-10 enrichment of the solution shall be determined anytime boron is added to the solution.

3.1 #52

(LAK 93-14R1)

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.1 REVISED PAGES

2A: MARKUP OF ITS

2B: NOT USED

(LAR 93-14R1)

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore ~~[secondary~~ primary containment} to OPERABLE status. (B1)	1 hour
	(B1) ~~AND~~		
	~~D.3~~	~~Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.~~	~~1 hour~~
	~~AND~~		
	~~D.4~~	~~Initiate action to restore one isolation valve and associated instrumentation to OPERABLE status in each [secondary containment] penetration flow path not isolated.~~ (P4)	~~1 hour~~
	AND		
	D.3	Initiate action to close one door in each primary containment air lock. ~~except during entry and exit under administrative control.~~ (P4)	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		

(continued)

- - - NOTE - - -
Entry and exit is permissible under administrative control.
- - - - - - - -

3.1 #60E

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		
	E.3	Initiate action to restore ~~[secondary containment]~~ primary containment to OPERABLE status. B1	1 hour
	~~AND~~		
B1	~~[E.4~~	~~Initiate action to restore one SGT subsystem to OPERABLE status.~~	~~1 hour]~~
	~~AND~~		
P4	~~E.5~~	~~Initiate action to restore one isolation valve and associated instrumentation to OPERABLE status in each [secondary containment] penetration flow path not isolated.~~	~~1 hour~~
	AND		
P4	E.4	Initiate action to close one door in each primary containment air lock, ~~except during entry and exit under administrative control.~~	1 hour

- - - NOTE - - -
Entry and exit is permissible under administrative control.
- - - - - - - -

3.1
#60E

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued) 3.1 #60k	A.2	~~-------NOTE-------~~ ~~Not applicable when less than or equal to the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS).~~ ~~------------------~~ Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS)
	AND		
	A.3	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	~~B.1~~	~~Disarm the associated CRD.~~	~~2 hours~~ C4
	~~AND~~		
	~~B.2~~	Be in MODE 3.	12 hours 3.1 #60j

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4 B1 a. No more than ~~[12]~~ 10 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

P3 b. No ~~more than 2~~ OPERABLE control rod~~s~~ that ~~are~~ is "slow" shall occupy ~~adjacent~~ a locations adjacent to another OPERABLE control rod that is "slow" or a withdrawn control rod that is stuck.

3.1 #60A

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1 B1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ ~~[~~950~~]~~ psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND (continued)

LAR 93-14R1

3.1 REACTIVITY CONTROL SYSTEMS

3.1 #60B

3.1.6 Rod Pattern Control

B1 LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the ~~[~~banked position withdrawal sequence (BPWS)~~]~~.

B1 APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 20 ~~[10]~~% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with ~~[~~BPWS~~]~~. (B1)	A.1	---------NOTE--------- Affected control rods may be bypassed in Rod Action Control System (RACS) in accordance with SR 3.3.2.1.~~8~~ 9. -------------------- (P3) Move associated control rod(s) to correct position.	8 hours
	OR		
	A.2	Declare associated control rod(s) inoperable.	8 hours

(continued)

LAR 93-14R1

3.1 #60B

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B1 B. Nine or more OPERABLE control rods not in compliance with ~~[~~BPWS~~].~~	B.1	--------NOTE--------- Affected control rods may be bypassed in RACS in accordance with SR 3.3.2.1.~~8~~ 9 for insertion only. P3 -------------------- Suspend withdrawal of control rods.	Immediately
	AND		
	B.2	Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
B1 SR 3.1.6.1	Verify all OPERABLE control rods comply with ~~[~~BPWS~~].~~	24 hours

---------- NOTE ----------
The minimum required available solution volume is determined by the performance of SR 3.1.7.5.

3.1 #51 P3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is ~~[≥ [4530] gallons]~~. greater than or equal to the minimum required available solution volume	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is ~~within the limits of [Figure 3.1.7-1]~~ ≥ 45°F.	24 hours
ADD INSERT 22A HERE (P4)		
~~SR 3.1.7.3~~	~~Verify temperature of pump suction piping is within the limits of [Figure 3.1.7-1].~~ (P5)	~~24 hours~~
SR 3.1.7.4	Verify continuity of explosive charge.	31 days
SR 3.1.7.5	~~Verify the concentration of boron in solution is [within the limits of Figure 3.1.7-1].~~ Verify the available weight of Boron-10 is ≥ 143 lbs, and the percent weight concentration of sodium pentaborate in solution is ≤ 9.5 % by weight, and determine the minimum required available solution volume. (P5) (3.1 #51)	31 days **AND** Once within 24 hours after water or boron is added to solution **AND** Once within 24 hours after solution temperature is restored ~~within the limits of [Figure 3.1.7-1]~~ to ≥ 45°F.

B1 B1 3.1 #51 B1 e e e e e

(continued)

INSERT 22A

SURVEILLANCE	FREQUENCY
----------NOTE---------- Sodium Pentaborate Concentration (C), in weight percent, is determined by the performance of SR 3.1.7.5. Boron-10 enrichment (E), in atom percent, is determined by the performance of SR 3.1.7.9. ---------- Verify that the SLC System satisfies the following equation: (C)(E) ≥ 413	3.1 # 52 31 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~[SR 3.1.7.10~~	~~Verify sodium pentaborate enrichment is ≥ [60.0] atom percent B-10.~~	~~Prior to addition to SLC tank]~~
SR 3.1.7.9	Determine Boron-10 enrichment of the solution.	Once within 24 hours after Boron is Added to the solution

B1

B1

e

3.1-52

LAR 93-14 R1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

3.1 #60B

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern ~~Control~~ Control"). Also, SDM is assumed as an initial condition for the control rod removal error during a refueling accident (Ref. 4). The analysis of this reactivity insertion event assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

C10 C2

The capability of inserting the control rods ~~ensures~~ provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the ~~strongest~~ highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

3.1 #60B

The control rods also protect the fuel from damage that could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLGHR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of the NRC Policy Statement.

LCO

OPERABILITY of an individual control rod is based on a combination of factors, primarily the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion

(continued)

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BASES

ACTIONS

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

3.1 #60j

Repeat as INSERT B16A

C7

2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. The control rod can be isolated from scram by isolating the hydraulic control unit from scram and normal ~~insert and withdraw~~ drive pressure, yet still maintain cooling water to the CRD.

3.1 #60F

3.1 #60K

Monitoring of the insertion capability for each withdrawn control rod must also be performed within 24 hours. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. The allowed Completion Time of 24 hours provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 ~~is modified by a Note that states the requirement is not applicable when~~ ~~below~~ the actual low power setpoint (LPSP) of the rod pattern controller (RPC), since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RPC (LCO 3.3.2.1, "Control Rod Block Instrumentation").

C6

THERMAL POWER is less than or equal to

has a modified time zero Completion Time. The 24 hour Completion Time for this Required Action starts when the withdrawn control rod is discovered to be stuck and THERMAL POWER is greater than

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with the

(continued)

BASES

ACTIONS

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

postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 7).

B.1 and B.2 C7 ~~INSERT B1.A from Page B3.1-15~~

3.1 #60i

With two or more withdrawn control rods stuck, ~~the stuck control rod should be isolated from scram pressure within 1 hour~~ ~~and~~ 2 hour the plant brought to MODE 3 within 12 hours. ~~The allowed Completion Time of 1 hour is acceptable, considering the low probability of a CRDA during this interval.~~ C4 The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RACS if required to allow insertion of the inoperable control rods and continued operation. SR 3.3.2.1.~~8~~ 9 provides additional requirements when the control rods are bypassed to ensure compliance with the CRDA analysis. P4

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide

(continued)

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BASES

ACTIONS (P3) E.1 ~~F.1~~ (continued)

inoperable control rods exist, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when (P5) operating above 20% ~~10%~~ RTP (i.e., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

3.1 #606 control rod

The position of each control rod must be determined, to ensure adequate information on control rod position is available to the operator for determining control rod ~~CRD~~ OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

SR 3.1.3.2 and SR 3.1.3.3

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to ~~below~~ the actual LPSP of the RPC since the notch insertions may not be compatible with the requirements

(C6) THERMAL POWER is less than or equal to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.5 (continued)

performed anytime a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.

P1

B1

2. U~~F~~SAR, Section ~~[~~4.3.2.5.5~~]~~.
3. U~~F~~SAR, Section ~~[~~4.6.1.1.2.5.3~~]~~.
4. U~~F~~SAR, Section ~~[~~5.2.2.2.3~~]~~.
5. U~~F~~SAR, Section ~~[~~15.4.1~~]~~.
6. U~~F~~SAR, Section ~~[~~15.4.9~~]~~.

3.1 #30

B1

7. NEDO-21231, "Banked Position Withdrawal Sequence," ~~[~~Section 7.2, January 1977.~~]~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 950 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL during the analyzed limiting power transient. Below 950 psig, the scram function is assumed to perform during the control rod drop accident (Ref. 6) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

3.1 #60B

Control rod scram times satisfy Criterion 3 of the NRC Policy Statement.

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met. To account for single failure and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin to allow up to 7.5% of the control rods (e.g., 145 [redacted] x 7.5% = 10 [redacted]) to have scram times that exceed the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times.

B1

(continued)

LAR 93-14R1

BASES

LCO (continued)

To ensure that local scram reactivity rates are maintained within acceptable limits, no ~~more than two of the allowed~~ "slow" control rod~~s~~ may occupy ~~adjacent~~ locations. Adjacent to another "slow" control rod or adjacent to a withdrawn stuck control rod.

P3 3.1 #60A

Table 3.1.4-1 is modified by two Notes, which state control rods with scram times not within the limits of the Table are considered "slow" and that control rods with scram times > ~~[7]~~ seconds are considered inoperable as required by SR 3.1.3.4.

B1

¶ INSERT B23A C6

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are ~~only allowed~~ not able to be withdrawn ~~under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide~~ adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

C3 INSERT B23B

ACTIONS

A.1

C7 INSERT B23C

When the requirements of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated (i.e., charging valve closed), the influence of the CRD pump head does not affect the single control rod scram times. During a full

(continued)

LAR 93-14R1

3.1 #38 a high probability of

INSERT B25A

The limits for reactor pressures < 950 psig are established based on ~~the expected relationship to~~ a high probability of meeting the acceptance criteria at reactor pressures ≥ 950 psig. Limits for ≥ 950 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7 second limit of Table 3.1.4-1 Note 2, the control rod can be declared OPERABLE and "slow."

LAR 93-14R1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 1, 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and, therefore, the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). Also, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

3.1 #60B

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod scram accumulators satisfy Criterion 3 of the NRC Policy Statement.

LCO

The OPERABILITY of the control rod scram accumulators is required to ensure that adequate scram insertion capability exists when needed over the entire range of reactor pressures. The OPERABILITY of the scram accumulators is based on maintaining adequate accumulator pressure.

APPLICABILITY

In MODES 1 and 2, the scram function is required for mitigation of DBAs and transients and, therefore, the scram accumulators must be OPERABLE to support the scram function. In MODES 3 and 4, control rods are ~~only allowed~~ not able to be withdrawn ~~under Special Operations LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown," and LCO 3.10.4, "Single Control Rod Withdrawal—Cold Shutdown," which provide~~ since the reactor mode switch is in Shutdown and a control rod block is applied. This provides adequate requirements for control rod scram accumulator OPERABILITY under these conditions. Requirements for scram accumulators in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

C3

C3

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod scram accumulator. This is acceptable since the Required Actions for each Condition provide appropriate compensatory action for each ~~inoperable~~ affected control rod. Complying with the Required Actions may allow for continued operation and subsequent ~~inoperable~~ affected control rods governed by subsequent Condition entry and application of associated Required Actions.

3.1 #60H

C2

A.1 and A.2

P3

With one control rod scram accumulator inoperable and the reactor steam dome pressure ≥ ~~900~~ 600 psig, the control rod may be declared "slow," since the control rod will still scram at the reactor operating pressure but may not satisfy the required scram times in Table 3.1.4-1 ~~in the LCO~~. Required Action A.1 is modified by a Note, which clarifies

C5

(continued)

LAR 93-1421

3.1 #60B

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod pattern controller (RPC) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted ~~to [10]% RTP~~ up to the low power setpoint (LPSP). (B1) The sequences effectively limit the potential amount of reactivity addition that could occur in the event of a control rod drop accident (CRDA).

(P1) This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References ~~1, 2, and 3~~ 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References ~~1, 2, and 3~~ 1 and 2. (P1) CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RPC (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage, which could result in undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage, which would result in release of radioactivity (Refs. ~~5 and 6~~ 4 and 5). Generic evaluations (Refs. ~~1 and 2~~ 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. ~~8~~ 7) and the calculated offsite doses will be well within the required limits (Ref. 5). (P1)

(continued)

LAR 93-14R1

3.1 #60B

BASES

APPLICABLE SAFETY ANALYSES (continued) P1 P4

Control rod patterns analyzed in Reference ~~1~~ 2 follow the banked position withdrawal sequence (BPWS) described in Reference ~~9~~ 8. The BPWS is applicable from the condition of all control rods fully inserted to ~~[illegible]~~ 20 RTP (Ref. ~~1~~ 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. ~~8~~ 9) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods. P1 P1 P1

Rod pattern control satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY P4 P1

In MODES 1 and 2, when THERMAL POWER is ≤ ~~[illegible]~~ 20 RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > ~~[illegible]~~ 20 RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. ~~1~~ 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

LAR 93-14R1

3.1
#60B

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, action may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive
B1 20 cooling water transient, leaking scram valves, or a power reduction to ≤ [10]% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the control rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note, which allows control rods to be bypassed in Rod Action Control System (RACS) to allow the affected control rods to be returned to their correct position. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3; LCO 3.1.4, "Control Rod Scram Times"; and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note

(continued)

LAR 93-14R1

3.1 #60B

BASES

ACTIONS

B.1 and B.2 (continued)

that allows the affected control rods to be bypassed in RACS in accordance with SR 3.3.2.1.~~8~~ 9 to allow insertion only. P3

With nine or more OPERABLE control rods not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the reactor mode switch in shutdown, the reactor is shut down, and therefore does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency, ensuring the assumptions of the CRDA analyses are met. The 24 hour Frequency of this Surveillance was developed considering that the primary check of the control rod pattern compliance with the BPWS is performed by the RPC (LCO 3.3.2.1). The RPC provides control rod blocks to enforce the required control rod sequence and is required to be OPERABLE when operating at ≤ ~~10%~~ 20% RTP. P4

REFERENCES

P1

~~1. Current Cycle Safety Analysis.~~ NEDE-24011-P-A, "GE Standard Application for Reactor Fuel, GESTAR II" (latest approved revision).

1. ~~2.~~ ~~"Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners Group, July 1987.~~ P2

2. ~~3.~~ USAR, Section 15.4.9. P5

3. ~~4.~~ NUREG-0979, "NRC Safety Evaluation Report ~~for~~ Related To The Final Design Approval of The GESSAR II BWR/6 Nuclear Island Design, Docket No. 50-447," Section 4.2.1.3.2, April 1983.

4. ~~5.~~ NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

(continued)

BASES

REFERENCES (continued)

5. 10 CFR 100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

P1

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," [January 1977.]

P3

INSERT B41A

3.1 #52

The requirements of 10 CFR 50.62 (Ref. 1) are met by the use of a sodium pentaborate solution enriched in the B-10 isotope (65 atom percent B-10). ~~Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.~~ P4 e

SR 3.1.7.3 determines whether the sodium pentaborate concentration, in conjunction with the B-10 enrichment, is within limits to meet the requirements of 10 CFR 50.62 (Ref. 1).

SR 3.1.7.5 ensures that the parameters used in the determination of sodium pentaborate concentration are within limits. This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure the proper weight of B-10 exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to establish that the weight of B-10 is within the specified limits. This Surveillance must be performed anytime the solution temperature is restored to ≥ 45°F, to ensure no significant boron precipitation occurred.

The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

3.1 #52 The available solution volume is the solution volume above the pump suction penetration into the storage tank.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.1.7.8 and SR 3.1.7.9 (continued)~~

P10 ~~temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.9, it is determined that the temperature of this piping has fallen below the specified minimum, this Surveillance must be performed once within 24 hours after the piping temperature is restored within the limits of Figure 3.1.7-1.~~

~~SR 3.1.7.10~~

P9 ~~Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.~~

REFERENCES

1. 10 CFR 50.62.

P1 2. U~~F~~SAR, Section [9.3.5.3]. B1

SR 3.1.7.9

3.1#52

P9

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the sodium pentaborate solution to determine the actual B-10 enrichment must be performed once within 24 hours after boron is added to the solution in order to ensure that the B-10 enrichment is adequate. Enrichment testing is only required when boron addition is made since enrichment change cannot occur by any other process.

ACTIONS (continued)

for each Condition provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

Since the SDV is still isolable, the affected SDV line may be opened. This allows accumulated water in the line to be drained to preclude a Reactor scram on SDV high level.

3.1 #60C

A.1

in C1

When one SDV vent or drain valve is in operable in one or more lines, the valves must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring during the time the valve(s) are inoperable ~~and the lines not isolated~~. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion ~~will~~ may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

C4

C1

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining of the SDV when a line is isolated. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line ~~(from leakage past the CRDs)~~ to be drained, to preclude a reactor scram on SDV high level. This is acceptable, since the ~~valves are operated from the control room and~~ administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

C4

C2

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikelihood of significant CRD seal leakage.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO

(continued)

LAR 93-14R1

RIVER BEND

SECTION 3.2

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.2 REVISED PAGES

1A: MARKUP OF CTS

1B: NOT USED

1C: NOT USED

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-14R1

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LCO 3.2.4

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-high scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships: R1

a. Two Recirculation Loop Operation

Trip Setpoint	Allowable Value
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$ R1

3.2.4.a 3.2.4.b

b. Single Recirculation Loop Operation

Trip Setpoint	Allowable Value
$S \leq (0.66W + 42.7\%)T$	$S \leq (0.66W + 45.7\%)T$
$S_{RB} \leq (0.66W + 36.7\%)T$	$S_{RB} \leq (0.66W + 39.7\%)T$ R1

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.

LA3

$T = \frac{3 \times FRTP + 1}{4 \times CMFLPD}$ provided $CMFLPD \leq 0.6 \times FRTP + 0.4$, otherwise

$T = \frac{FRTP}{CMFLPD}$

T is applied only if less than or equal to 1.0.

FRTP is the FRACTION OF RATED THERMAL POWER.
CMFLPD is the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY.

A1 APPLICABILITY: ≥ ~~OPERATIONAL CONDITION 1, when~~ THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

3.2 #76

ACTION:

R1 With the APRM flow biased simulated thermal power-high scram trip setpoint ~~and/or~~ the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value * within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

Cond A Cond B LA1

*With T < 1.0, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the adjusted APRM readings result in a calculated T ≥ 1.0 when the APRM reading is substituted for FRTP, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

3.2.4.c LA2

(LAR 93-14R1)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 4.2.2 The FRTP and CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required: R1

a. At least once per 24 hours,

L1 b. Within 12 hours after ~~completion of a~~ THERMAL POWER ~~increase of at least 15% of~~ ≥ 25% RATED THERMAL POWER, and

3.2.4.2 c. Initially and at least once per 12 hours when the reactor is operating with T ≤ 1.0.

A2 d. The provisions of Specification 4.0.4 are not applicable.

3.2 #76

(LAR 93-14R1)

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.2 REVISED PAGES

2A: MARKUP OF ITS

2B: NOT USED

(LAR 93-14R1)

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints ~~(Optional)~~ B1

LCO 3.2.4 3.2 #76

a. T ~~MFLPD~~ shall be ≥ 1.0 ~~less than or equal to Fraction of RTP~~; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the adjusted APRM readings ~~are ≥ 100% times MFLPD.~~ result in a calculated T ≥ 1.0 when the APRM reading is substituted for FRTP.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------ NOTE ------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4, Item b or c requirements. ------ Verify ~~MFLPD~~ T is ~~within limits~~ ≥ 1.0. 3.2 #76	Once within 12 hours after ≥ 25% RTP **AND** 24 hours thereafter

E2

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	------ NOTE ------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4, Item a requirements. ------ Verify APRM setpoints or gains are adjusted for the calculated ~~MFLPD~~ T. 3.2 #76	12 hours

BASES

LCO (continued)

recirculation loops operating, the limit is determined by ~~multiplying~~ the ~~smaller of the MAPFAC_f and MAPFAC_p factors~~ e ~~times the~~ exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by ~~the smallest of MAPFAC_f, MAPFAC_p,~~ e ~~and 0.86, where 0.86 has been~~ determined by a specific single recirculation loop analysis ~~(Ref. 3)~~.

P6 a value P5

APPLICABILITY

P11 The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations ~~(Ref. 4)~~ and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ~~≤~~ < 25% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.

3.2 #73

ACTIONS

A.1

C6 If any APLHGR exceeds the required limit~~s~~, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limit(s) and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

C6 C6

B.1

C6 If the APLHGR cannot be restored to within its required limit(s) within the associated Completion Time, the plant must

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency ~~(Refs. 3, 4, and 5)~~. Flow dependent MCPR limits (MCPR$_f$) are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 5 ~~6~~) ~~and the multichannel thermal hydraulic code (Ref. 7)~~. $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual ~~and Non Loop Manual~~ operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. ~~Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.~~

P5 P5 P7 P7 3.2 #67

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref 6 ~~9~~). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow ~~$MAPFAC_p$~~ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

P5 $MCPR_p$ ... C10

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed ~~with the flow control valve in its minimum position~~ and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

C3

(continued)

31

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENC
SR 3.2.4.1	Verify ~~MFLPD~~ T is ~~within limits~~ ≥ 1.0	Once within 12 hours af ≥ 25% RTP **AND** 24 hours thereafter

3.2 #76

C2

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

Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4, Item b or c requirements.

SURVEILLANCE		FREQUENCY
SR 3.2.4.2	---------- NOTE ---------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4, Item a requirements. ------------------ Verify APRM setpoints or gains are adjusted for the calculated T ~~MFLPD~~.	12 hours

3.2 #76

BASES

LCO (continued)

recirculation loops operating, the limit is determined by ~~multiplying~~ the ~~smaller of the MAPFAC, and MAPFAC, factors times the~~ exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by ~~the smallest of MAPFAC, MAPFAC, and 0.86, where 0.86 has been~~ determined by a specific single recirculation loop analysis ~~(Ref. 7)~~.

P6 a value P5

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations ~~(Ref. 9)~~ and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ~~≤~~ < 25% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.

P11 3.2 #73

ACTIONS

A.1

C6 C6 C6

If any APLHGR exceeds the required limit~~s~~, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limit~~s~~ and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

C6

If the APLHGR cannot be restored to within its required limit~~s~~ within the associated Completion Time, the plant must

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency ~~(Refs. 2, 4, and 5)~~. Flow dependent MCPR limits (MCPRf) are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 5 ~~6~~) ~~and the multichannel thermal hydraulic code (Ref. 7)~~. $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual ~~and Non Loop Manual~~ operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. ~~Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.~~

P5 P5 P7 P7

(MCPRf) 3.2 #67

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 6 ~~8~~). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow ~~MAPFAC~~ MCPRp operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level. C10

P5

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed ~~with the flow control valve in its minimum position~~ and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

C3

(continued)

LAR 93-1481

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND C7

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that ~~assure~~ assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

INSERT B 12A

3.2 #76

C7

~~The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP) where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:~~

$$\frac{MFLPD}{FRTP} > 1,$$

~~indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR and MCPR.~~

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

(continued)

INSERT B12A

3.2 #76

The condition of high power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at Rated Thermal Power (RTP). This ratio is presented as a APRM setpoints T-factor (T). At any off-rated power conditions T is equal to the Fraction of Rated Thermal Power (FRTP) divided by the Core Maximum Fraction of Limiting Power Density (CMFLPD). FRTP is the measured thermal power divided by the RTP. CMFLPD is the limiting linear heat generation rate (LHGR) divided by the rated LHGR limit. High power peaking exists when:

$$\frac{\text{FRTP}}{\text{CMFLPD}} \leq 1.0$$

As power is reduced with the design power distribution maintained, CMFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the CMFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gains are adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM gains or flow biased scram trip setpoints. Adjusting the APRM gains or setpoints is equivalent to maintaining CMFLPD less than or equal to FRTP.

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With T < 1.0, rather than adjusting the APRM setpoints,

BASES

may be adjusted such that

3.2 #76

LCO (continued)

c. ~~Increasing~~ the APRM gains ~~to cause~~ the adjusted APRM ~~to read greater than 100(%) times MFLPD~~. This Condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

readings result in a calculated T ≥ 1.0 when the APRM reading is substituted for FRTP

INSERT B15A

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting the APRM gain or setpoints is equivalent to maintaining ~~MFLPD less than or equal to FRTP~~ T ≥ 1.0 as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

3.2 #76

APPLICABILITY

The ~~MFLPD~~ T factor limit, APRM gain adjustment, or APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2 sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the plant is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while ~~the MFLPD has exceeded FRTP~~ T < 1.0, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)

BASES

ACTIONS

A.1 (continued)

T

3.2 #76

may be reduced. Therefore, prompt action should be taken to restore ~~the MFLPD~~ to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If the APRM gain or setpoints cannot be restored to within their required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

3.2 #76

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

T

These SRs are

C2

~~MFLPD and, assuming MFLPD is greater than FRTP, the~~

of SR 3.2.4.1

3.2 #76

T and

3.2 #76

~~The MFLPD~~ is required to be calculated ~~every 24 hours~~ and compared to ~~FRTP or~~ APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. ~~This SR is~~ required only to determine the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. ~~assuming MFLPD is greater than FRTP.~~ The 24 hour Frequency is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if ~~MFLPD is less than or equal to FRTP~~ T ≥ 1.0. When ~~MFLPD is greater than FRTP~~ T < 1.0, more rapid changes in power distribution are typically expected.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.

B1 2. UFSAR, ~~Section [ ]~~ Chapter 4, Appendix 4B.

P1 3. UFSAR, ~~Section [ ]~~ Chapter 15, Appendix 15B.

3.2 #76 4. NRC Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors."

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RIVER BEND

SECTION 3.3

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.3 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

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ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3 ~~8~~. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (continued)	A10			
i. Main Steam Line Tunnel Temperature Timer	2	1	1. 2, 3	~~27~~ F/H
j. RHR Equipment Room Ambient Temperature - High	2	1	1, 2, 3	~~27~~ F/H
k. RHR Equipment Room Δ Temperature - High	2	1	1, 2, 3	~~27~~ F/H
l. RHR/RCIC Steam Line Flow - High	2	1	1, 2, 3	~~27~~ F/H
m. Drywell Pressure - High	$3^{(g)}$	1	1, 2, 3	~~27~~ F/H
n. Manual Initiation	$2^{(k)}$	1	1, 2, 3	~~28~~ G/H LB
5 ~~6~~. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	5, 14	2	1, 2, 3	~~30~~ F/H
b. RHR Equipment Area Δ Temperature - High	5, 14	2	1, 2, 3	~~30~~ F/H
c. Reactor Vessel Water Level - Low, Level 3	5, 14	2	1, 2, 3	~~30~~ F/H
d. Reactor Vessel Water Level - Low Low Low, Level 1	10	2	m5 1, 2, 3 4, 5	~~30~~ F/H J m4

3.3.6.1 #21,22

3.3.6.T #45

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<SR 3.3.6.1.7> Table 3.3.6.1-1 <LCO 3.3.6.1 LCO 3.3.6.2>

~~TABLE 3.3.2-3~~

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	3.3.6.1 #38
a. Reactor Vessel Water Level - Low Low Level 2	7- $\leq 10^{(a)}$ LA1
b. Drywell Pressure - High	7- $\leq 10^{(a)}$
c. Containment Purge Isolation Radiation - High$^{(b)}$	NA L18
2. MAIN STEAM LINE ISOLATION	LA1
a. Reactor Vessel Water Level - Low Low Low Level 1	7- ≤ 1.0 */$\leq 10^{(a)}$**
L5 b. Main Steam Line Radiation - High$^{(b)}$	≤ 1.0 */$\leq 10^{(a)}$**
c. Main Steam Line Pressure - Low	7- ≤ 1.0 */$\leq 10^{(a)}$**
3.3.6.1 #12 d. Main Steam Line Flow - High	7- ≤ 0.5 */$\leq 10^{(a)}$**
e. Condenser Vacuum - Low	NA
R1 f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Area Temperature - High (Turbine Bldg)	NA 3.3.6.2 #35
3. SECONDARY CONTAINMENT ISOLATION Table 3.3.6.2-1 <SR 3.3.6.2.6>	
a. Reactor Vessel Water Level - Low Low Level 2	6- $\leq 10^{(a)}$ LA1 L18
b. Drywell Pressure - High	6- $\leq 10^{(a)}$
c. Fuel Building Ventilation Exhaust Radiation - High$^{(b)}$	NA
d. Reactor Building Annulus Ventilation Exhaust Radiation - High$^{(b)}$	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION R1	3.3.6.1 #38
a. Δ Flow - High	7- $\leq 10^{(a)}$## LA1
b. Δ Flow Timer	NA
R1 c. Equipment Area Temperature - High	NA L18
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$ LA1
R1 f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	7- $\leq 10^{(a)}$### LA1
b. RCIC Steam Line Flow-High Timer	NA
c. RCIC Steam Supply Pressure - Low	7- $\leq 10^{(a)}$ LA1
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
e. RCIC Equipment Room Ambient Temperature - High	NA
R1 f. RCIC Equipment Room Δ Temperature - High	NA
g. Main Steam Line Tunnel Ambient Temperature - High	NA
h. Main Steam Line Tunnel Δ Temperature - High	NA

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LCO 3.3.6.1
LCO 3.3.6.2

~~TABLE 3.3.2-3 (Continued)~~

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
i. Main Steam Line Tunnel Temperature Timer	NA
j. RHR Equipment Room Ambient Temperature - High	NA
~~k. RHR Equipment Room Δ Temperature - High~~	~~NA~~
l. RHR/RCIC Steam Line Flow - High	NA
m. Drywell Pressure - High	NA
n. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
~~b. RHR Equipment Area Δ Temperature - High~~	~~NA~~
c. Reactor Vessel Water Level - Low Level 3	7 ≤ 10(a)
d. Reactor Vessel Water Level - Low Low Low Level 1	7 ≤ 10(a)
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
7. MANUAL INITIATION	NA

R1 3.3.6.1 #12 R1 3.3.6.1 #38 L18 LA1 L18

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays. SR 3.3.6.1.1 Note SR 3.3.6.2.1 Note R1

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel. LA1

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed. L18

LA2 ~~**Isolation system instrumentation response time for associated valves except MSIVs.~~

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time ~~shown in Tables 3.6.4-1 and 3.6.5.3-1~~ for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve. NO A10

LA1

##Time delay of 45-47 seconds.

###Time delay of 3-13 seconds.

Table 3.3.2.1-1

~~TABLE 4.3.6-1~~

See Markup of Table 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	SR 3.3.2.1.X CHANNEL FUNCTIONAL TEST	SR 3.3.2.1.X CHANNEL CALIBRATION (a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	S(f)	S/U(b)(e), ~~M~~ Q(e)	SA	1, 2
b. High Power Setpoint	S(f)	S/U(b)(e), ~~M~~ Q(e)	SA	1**
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	S/U(b), ~~M~~ Q	SA(g)	1
b. Inoperative	NA	S/U(b), ~~M~~ Q	NA	1, 2, 5
c. Downscale	NA	S/U(b), ~~M~~ Q	SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b), ~~M~~ Q	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	SA	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W	NA	2, 5
b. Upscale	NA	S/U(b), W	SA	2, 5
c. Inoperative	NA	S/U(b), W	NA	2, 5
d. Downscale	NA	S/U(b), W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	~~M~~ Q	R#	1, 2, 5*
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b), ~~M~~ Q	SA(g)	1

LA1 L2 Note LA2 LS A4 L2 LA2 LA2 3.3.2.1 #4 R1

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<LCO 3.3.2.1>

REACTIVITY CONTROL SYSTEMS

<LCO 3.3.2.1>

ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

SR 3.3.2.1.9 ~~c.~~ The position and bypassing of an inoperable control rod(s) is verified by a second licensed operator or other technically qualified member of the unit technical staff.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1 ~~4.1.4.2~~ The RPCS shall be demonstrated OPERABLE by verifying the OPERABILITY of the:

SR 3.3.2.1.5 ~~a.~~ Rod pattern controller functions when THERMAL POWER is less than the low power setpoint by selecting and attempting to move an inhibited control rod: LA2

SR 3.3.2.1.3 ~~1.~~ After withdrawal of the first insequence control rod or gang for each reactor startup. L2 NOTE

SR 3.3.2.1.4 ~~2.~~ As soon as the rod inhibit mode is automatically initiated at the RPCS low power setpoint, 20 + 15, - 0% of RATED THERMAL POWER, during power reduction. LA1

3. The first time only that a banked position, N1, N2, or N3, is reached during startup or during power reduction below the RPCS low power setpoint. L2

SR 3.3.2.1.5 ~~b.~~ Rod withdrawal limiter functions when THERMAL POWER is greater than or equal to the low power setpoint by selecting and attempting to move a restricted control rod in excess of the allowable distance: LA2

3.3.2.1 #15 L6

SR 3.3.2.1.1 NOTE
SR 3.3.2.1.2 NOTE ~~1.~~ As each power range above the RPCS low power setpoint is entered during a power increase or decrease. L2

SR 3.3.2.1.1
SR 3.3.2.1.2 ~~2.~~ At least once per ~~31~~ 92 days while operation continues within a given power range above the RPCS low power setpoint. A4

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.3.1.2.5 b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and (L4)

2. At least once per 7 days. (L6) - - - 3.3.1.2 #10

c. Verifying that the channel count rate is at least 0.7 cps*:

1. Prior to control rod withdrawal, (L4)

SR 3.3.1.2.4 2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and (L4)

3. At least once per 24 hours, except that:

(M3) - - - a. During fuel unloading, the required count rate may be permitted to be less than 0.7 cps*.

SR 3.3.1.2.4 NOTE (L1) - - - b. Prior to and during fuel loading, until sufficient fuel has been loaded to maintain at least 0.7 cps*, the required count rate may be achieved by:

a) Use of portable external source, or

b) Loading up to 2 fuel assemblies in cells containing inserted control rods around an SRM.

(L2) d. Verifying within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn## that the RPS circuitry "shorting links" have been removed unless adequate shutdown margin has been demonstrated per Specification 3.1.1.

INSERT SR 3.3.1.2.6 (A4)

3.3.1.2 #10

<SR 3.3.1.2.4

*Provided signal to noise ratio ≥ 2, otherwise use 3.0 cps.

(A3) ##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.3.1 - RPS INSTRUMENTATION

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.4 Applicability has been modified to only require RPS functions to be OPERABLE in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and therefore are not required to be OPERABLE with the capability to scram. Provided all rods otherwise remain inserted, the RPS functions serve no purpose and are not required. In this condition the required SHUTDOWN MARGIN (LCO 3.1.1) and the required one-rod-out interlock (LCO 3.9.2) ensure no event requiring RPS will occur. The ACTIONS for inoperable equipment in MODE 5 are also revised to be consistent with the proposed Applicability. Since all control rods are required to be fully inserted during fuel movement (LCO 3.9.3), the proposed applicable conditions cannot be entered while moving fuel. The only possible core alteration is control rod withdrawal which is adequately addressed by the proposed ACTION.

L.5 The main steam line radiation monitor (MSLRM) scram function has been removed from the Technical Specifications based on the guidelines provided by General Electric NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor." This technical report provided the results of generic evaluations which indicated the MSLRMs are unnecessary to ensure compliance with the radiation dose guidelines of 10 CFR Part 100. EOI has confirmed that the analyses presented in NEDO-31400A are applicable to RBS and bound the results of analyses for RBS. Additionally, the MSLRM is not credited for a reactor scram initiation for any design basis event. Finally, the reliability assessment of the elimination of the scram function on reactivity control failure frequency and core damage frequency indicate a net improvement in safety.

3.3.1.1 #13

EOI confirms that, upon disabling of the MSLRM scram function, the conditions identified in the NRC's SER for NEDO-31400A will be implemented as identified below. With the implementation of the technical report (NEDO-31400A) guidelines, the main steam line radiation monitor and offgas radiation monitor alarm setpoints will be standardized above the nominal background dose rate to provide the indication of need for a prompt sample of the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective action. Any significant increases in the levels of radioactivity in the main steam lines will be expeditiously controlled (by procedure) to limit both occupational doses and environmental releases.

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DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

TECHNICAL CHANGE - MORE RESTRICTIVE
(continued)

M.2 This comment number is not used for this station.

M.3 The addition of Condition H provides specific guidance of the ACTIONS if the current ACTION cannot be met. Currently this would require entry into LCO 3.0.3 which allows one hour to initiate the shutdown; under Condition H, shutdown is required, but the one hour to initiate the shutdown is omitted.

3.3.6.1 #45

M.4 An additional requirement is included as Required Action J.3 that requires action be initiated to restore the primary containment boundary if the RHR Shutdown Cooling System is not isolated or the instrumentation is not restored to OPERABLE status.

M.5 The proposed Applicability has added a requirement for Reactor Vessel Water Level - Low Level 3 instrumentation to be OPERABLE for the RHR System isolation logic. Since this is not presently a requirement, this change is more restrictive.

M.6 This comment number is not used for this station.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 Testing of the response time is provided by a specific SR and is an integral part of the OPERABILITY of certain instrumentation channels. Details of the methods for performing this and other surveillances and Required Actions are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the USAR. Additionally, changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications.

LA.2 System design and operational details have been relocated to the Bases and procedures. Trip setpoints are an operational detail that is not directly related to the OPERABILITY of the instrumentation. The Allowable Value is the required limitation for the parameter and this value is retained. Details relating to system design and operation (e.g., commonality with RPS, bypasses, specific valves or valve groups affected, etc.) are also unnecessary in the LCO and have been relocated to the Bases and procedures. The design features and system operation are also described in the USAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications.

<LA.3 and LA.4 moved to this page due to addition of M.4 on page 15> (LAR 93-14R1)

DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

LA.3 This comment number is not used for this station.

LA.4 This comment number is not used for this station.

LB.1 This comment number is not used for this station.

LB.2 The note is clarified to provide direct indication of the intent of the current wording. Providing "at least one OPERABLE channel in the same trip system ... monitoring that parameter" is intended to assure that the trip capability of that function is maintained. However, it does not provide this assurance for all logic system designs. The proposed Note is based on previously conducted reliability analyses (NEDC-31677-P-A and NEDC-30851-P-A, Supplement 2).

"Specific"

L.1 The Frequency for the Channel Functional Test of the SLCS initiation logic is extended from every 6 months (i.e., every 92 days on a STAGGERED TEST BASIS) to 18 months. This function is manually initiated and the testing is very similar to the testing of the manual initiation function. The 18 months is based on the potential for an unplanned transient due to the loss of the reactor water cleanup system flow when the surveillance is performed with the reactor at power. In addition, operating experience at other plants has shown these components usually pass the surveillance when performed at this frequency.

L.2 This comment number is not used for this station.

L.3 This comment number is not used for this station.

L.4 The Required Action if the Required Action and associated Completion Time of Conditions A or B are not met is proposed to allow isolation of the affected main steam line(s). Some conditions may affect the isolation logic for only one main steam line. In these cases, it is not necessary to require a shutdown of the unit; rather, isolation of the affected line returns the system to a status where it can perform the remainder of its isolation function, and continued operation is allowed (although it may be at a reduced power level.)

DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.5 The main steam line radiation monitor (MSLRM) scram and MSL isolation functions have been removed from the Technical Specifications based on the guidelines provided by General Electric NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor." This technical report provided the results of generic evaluations which indicated the MSLRMs are unnecessary to ensure compliance with the radiation dose guidelines of 10 CFR Part 100. EOI has confirmed that the analyses presented in NEDO-31400A are applicable to RBS and bound the results of analyses for RBS. Additionally, the MSLRM is not credited for a reactor scram initiation or MSL isolation for any design basis event. Finally, the reliability assessment of the elimination of the scram function on reactivity control failure frequency and core damage frequency indicate a net improvement in safety.

3.3.6.1 #7

EOI confirms that, upon disabling of the MSLRM scram function, the conditions identified in the NRC's SER for NEDO-31400A will be implemented as identified below. With the implementation of the technical report (NEDO-31400A) guidelines, the main steam line radiation monitor and offgas radiation monitor alarm setpoints will be standardized above the nominal background dose rate to provide the indication of need for a prompt sample of the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective action. Any significant increases in the levels of radioactivity in the main steam lines will be expeditiously controlled (by procedure) to limit both occupational doses and environmental releases.

L.6 The ACTION to isolate all main steam lines is a sufficient ACTION with the referenced functions inoperable and will require being in MODE 2 to avoid a scram. The requirement to be in MODE 2 is therefore implicit and is deleted. The time allowed to isolate the associated main steam lines is extended from 6 hours to 12 hours. The additional time is provided to allow for more orderly power reduction. Since 36 hours are otherwise provided to exit the applicable MODES, this short extension is still well within any reliability or availability assumptions.

L.7 This comment number is not used for this station.

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DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.8 The manual initiation function is not assumed in any accident or transient analysis in the USAR and isolation of the penetration due to loss of this function within one hour is considered overly conservative. The time allowed to isolate the line is extended to 24 hours to provide time for adequate preparation to accommodate the effects of isolating the penetration. (Note that the time allowed to isolate the penetration was reduced from 48 hours to 24 hours for other manual initiation functions. See discussion of change M.1.)

L.9 This comment number is not used for this station.

L.10 This comment number is not used for this station.

L.11 Options are provided that would allow isolation of the affected lines, or for secondary containment, either operation of the SGTS or a declaration of an inoperable SGTS. These ACTIONS conservatively compensate for the inoperable status of the instrumentation through restoration of the single failure capability or through providing the required instrumentation actuation function. Therefore, providing this option does not impact safety.

L.12 This comment number is not used for this station.

L.13 This comment number is not used for this station.

L.14 This comment number is not used for this station.

L.15 The SLCS is not required during MODE 3 since no control rods can be withdrawn and adequate SHUTDOWN MARGIN prevents criticality under these conditions. This is consistent with the Applicability requirements for the SLCS.

L.16 This comment number is not used for this station.

3.3.6.1 #38

L.17 This comment number is not used for this station.

3.3.6.2 #35

L.18 The proposed change deletes primary containment isolation instrumentation and secondary containment isolation instrumentation response time testing where the required response time corresponds to the required diesel start time. The bases for this specification is also revised consistent with deleting this testing. These changes are revised based on guidance provided in Generic Letter 93-05, Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation, September 1993.

As described in the current Technical Specifications bases, LAR-93-14R1

DISCUSSION OF CHANGES
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

with exception of the Main Steam line Isolation Valves (MSIVs), individual sensor response times or the response times of the logic systems to which the sensors are connected are not addressed in the safety analysis. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable reactor coolant inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. Thus, the signal delay (sensor response) is concurrent with the 10 second diesel startup. Since typical response times are measured in fractions of a second, the chance is remote that a channel's response would degrade to the point where it exceeds the 10-second diesel start time without a noticeable failure. Therefore, the proposed changes are justified.

3.3.6.1
#38

3.3.6.2
#35

LAR 93-14RT

DISCUSSION OF CHANGES
CTS: 3.3.6 - CONTROL ROD BLOCK INSTRUMENTATION

3.3.2.1 #15

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.6 ITS SR 3.3.2.1.9 allows control rods to be bypassed and repositioned under the direction of a second licensed operator or other qualified member of the technical staff. This allowance is currently only applicable (per CTS 3.1.4.2 ACTION b) for insertion of the bypassed control rod. CTS Special Test Exception 3.10.2 allows a bypassed control rod to be withdrawn to its prior position to support special tests, including scram time tests. The ITS has expanded this flexibility to allowing the control rod to be bypassed with the positioning of the control rods controlled by other Specifications. Namely, LCO 3.1.4 does not allow slow or stuck control rods to occupy adjacent locations. With respect to the Control Rod Drop Accident (CRDA), which is only of concern below the Low Power Setpoint (LPSP) of the Rod Pattern Control System (RPCS), LCO 3.1.6 requires OPERABLE control rods to be in compliance with the Banked Position Withdrawal Sequence (BPWS) analysis while LCO 3.1.3 Condition D requires inoperable control rods to either comply with the BPWS analysis or be separated by at least two core cells. This ensures that the control rods remain within the patterns assumed for the CRDA analysis. With respect to the Rod Withdrawal Error (RWE) accident, which is only applicable above the LPSP of the RPCS, adequate separation of the control rods is assured by the thermal operating limits as discussed in CTS 3.1.3.1 DOC L.1.

As stated above, the ITS has expanded the flexibility for bypassing and moving bypassed control rods. Implicit in the requirement that movement of the bypassed control rods be performed under the direction of a second licensed operator or other technically qualified member of the staff is that the positioning be in conformance with applicable safety analysis. When operating below the LPSP of the RPCS, the applicable analysis is the CRDA. Compliance with this analysis is ensured by conformance with the generic BPWS analysis or a specific BPWS analysis for the evolution. Similarly, when operating above the LPSP of the RPCS, the applicable analysis is the Rod Withdrawal Error (RWE) analysis. Movement of bypassed control rods must be in conformance with the generic RWE analysis or with a special analysis to ensure that the conclusions of the RWE analysis remains supported. The above controls ensure that positioning and movement of bypassed control rods remain within the bounds of previous analysis.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.3.7.6 - SOURCE RANGE MONITORS

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.6 As identified in DOC R.1 for CTS 3.3.6, the requirements for the SRM Control Rod Block Functions have been Relocated from the Technical Specifications to licensee controlled documents as they do not satisfy the NRC Policy Statement technical specification screening criteria. As a result, as further supported by the Bases for ITS LCO 3.3.1.2, the function of the SRMs, as required by the ITS is to provide the operators with information on neutron flux levels at low power levels and during shutdown conditions, including CORE ALTERATIONS. The ITS does not require that the SRMs provide any trip functions, only indication. These functions are adequately tested during MODE 5 operations via SR 3.3.1.2.1 which provides for a CHANNEL CHECK at least once per 12 hours with adequate count rate (per SR 3.3.1.2.4) and by SR 3.3.1.2.7 which provides for a CHANNEL CALIBRATION at least once per 18 months. Since the SRMs do not provide any trip functions and there are no additional functional requirements for the SRM channels while in MODE 5 than there are in MODE 2, 3, or 4, performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days as provided by NUREG-1434 SR 3.3.1.2.6 provides sufficient assurance of the functional capability of the required SRM channels.

3.3.1.2
#10

10/4/94

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

3.3.6.1 #38

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.3.2 - ISOLATION ACTUATION INSTRUMENTATION

"L18" CHANGE

3.3.6.2 #35

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Individual sensor response times for the affected instruments are not addressed in the accident analysis. A 10 second time delay is assumed prior to isolation valve movement. Thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. Since typical response times are measured in fractions of a second, the chance is remote that a channel's response would degrade to the point where it exceeds the 10 second diesel start time without a noticeable failure. Therefore, this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Response time testing on the affected instruments is only intended to enhance system reliability and to monitor instrument channel response time trends. The response times are concurrent with the 10 second diesel startup. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The instrumentation response times are concurrent with the 10 second diesel generator startup time assumed in the accident analysis. The proposed changes do not impact the safety analysis or a margin of safety. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

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3.3.2.1
#15

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.3.6 - CONTROL ROD BLOCK INSTRUMENTATION

"L6" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will allow control rods to be bypassed to allow movement other than insertion. CTS 3/4.1.4.2 allows inoperable control rods to be bypassed to allow them to be inserted and disarmed. However, CTS 3.10.2 allows OPERABLE control rods to be bypassed to allow performance of scram time and other testing. Because the proposed change continues to require these movements to be performed in conformance with the control rod drop and rod withdrawal error analyses, as applicable, this change does not significantly increase the probability of a previously analyzed accident nor does it significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation (since it is currently allowed by CTS 3.10.2) and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change continues to require these movements to be performed in conformance with the control rod drop and rod withdrawal error analyses, as applicable.

10/4/94

3.3.1.2
#10

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.3.7.6 - SOURCE RANGE MONITORS

"L6" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will increase the Channel Functional Test interval for the Source Range Monitors (SRMs) while operating in MODE 5. There are no accidents that are initiated by the failure of the SRMs. Therefore, this change does not increase the probability of a previously analyzed accident. Further, the ITS does not require the SRMs to perform any trip functions, only indication (see CTS 3.3.6 comment R.1). These functions are assured by the performance of a Channel Check every 12 hours per SR 3.3.1.2.4 and a Channel Calibration per SR 3.3.1.2.7 every 18 months. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since, as stated above, the SRMs are not required to perform any trip functions, only provide indication. These functions are adequately assured by the performance of a Channel Check every 12 hours per SR 3.3.1.2.4 and a Channel Calibration per SR 3.3.1.2.7 every 18 months.

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ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.3 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.4	------------------NOTE------------------ Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. -- Verify count rate is: a. ≥ [3.0] cps ~~with a signal to noise ratio ≥ [2:1]~~, or b. ≥ [0.7] cps with a signal to noise ratio ≥ ~~[20:1]~~ 2:1. (P2)	12 hours during CORE ALTERATIONS **AND** 24 hours
~~SR 3.3.1.2.5~~	~~Perform CHANNEL FUNCTIONAL TEST.~~	~~7 days~~
SR 3.3.1.2.~~6~~ 5	------------------NOTE------------------ Not required to be performed until 12 hours after IRMs on Range 2 or below. -- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.2.~~7~~ 6	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. (C2) -- Perform CHANNEL CALIBRATION.	~~[18]~~ months

3.3.1.2 #10

Table 3.3.1.2-1 (page 1 of 1)
Source Range Monitor Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Source Range Monitor	$2^{(a)}$	~~4~~ 3 (31)	SR 3.3.1.2.1 SR 3.3.1.2.4 SR 3.3.1.2.~~6~~ 5 SR 3.3.1.2.~~7~~ 6
	3,4	2	SR 3.3.1.2.3 SR 3.3.1.2.4 SR 3.3.1.2.~~6~~ 5 SR 3.3.1.2.~~7~~ 6
	5	$2^{(b),(c)}$	SR 3.3.1.2.1 SR 3.3.1.2.2 SR 3.3.1.2.4 SR 3.3.1.2.5 SR 3.3.1.2.~~7~~ 6

3.3.12 #10

(a) With IRMs on Range 2 or below.

(b) Only one SRM channel is required to be OPERABLE during spiral offload or reload when the fueled region includes only that SRM detector.

(c) Special movable detectors may be used in place of SRMs if connected to normal SRM circuits.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.4 B1	------NOTE------ Not required to be performed until 1 hour after THERMAL POWER is ≤ ~~[10]~~ 20% RTP in MODE 1. ------ Perform CHANNEL FUNCTIONAL TEST.	[92] days B1
3.3.2.1 #4 SR 3.3.2.1.5 B1	Calibrate the low power setpoint trip units. The Allowable Value shall be ~~≥~~ > ~~[10]~~ 20% RTP and ~~≤ [35]% RTP~~. C11	~~184~~ 92 days C4
SR 3.3.2.1.6 B1 68.2%	Verify the RWL high power Function is not bypassed when THERMAL POWER is > ~~[70]%~~ RTP.	~~184~~ 92 days
SR 3.3.2.1.7	Perform CHANNEL CALIBRATION.	184 days C4
SR 3.3.2.1.~~7~~8 C4	------NOTE------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ------ Perform CHANNEL FUNCTIONAL TEST.	[18] months B1
SR 3.3.2.1.~~8~~9 3.3.2.1 #15	Verify the bypassing and movement of control rods required to be bypassed in Rod Action Control System (RACS) is in conformance with applicable analyses ~~by a second licensed operator or other qualified member of the technical staff~~.	Prior to and during the movement of control rods bypassed in RACS

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Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Rod Pattern Control System			SR 3.3.2.1.7 C4
a. Rod withdrawal limiter	~~[(a)]~~ a B1	2	SR 3.3.2.1.1 SR 3.3.2.1.6 SR 3.3.2.1.9
	~~[(b)]~~ a B1	2	SR 3.3.2.1.2 SR 3.3.2.1.5 SR 3.3.2.1.9
b. Rod pattern controller	1(c),2(~~c~~) b P4	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.5 SR 3.3.2.1.~~8~~9
2. Reactor Mode Switch – Shutdown Position	(d)	2	SR 3.3.2.1.~~7~~8

B1

(a) THERMAL POWER > ~~[70]~~ 68.2 % RTP.

(b) THERMAL POWER > [35] % RTP and ≤ ~~70%~~ 68.2 RTP. P4

(c) With THERMAL POWER ≤ ~~[10]~~ 20 % RTP.

(d) Reactor mode switch in the shutdown position.

3.3.2.1 #4

3.3.2.1 #15

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3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 ~~Initiate action in accordance with Specification 5.8.2.c.~~ P2 8 to prepare and submit a Special Report.	Immediately 3.3.3.1 #17
C. ~~---------NOTE--------- Not applicable to [hydrogen monitor] channels. ---------------------~~ P4 One or more Functions with two required channels inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
P4 ~~D. Two [required hydrogen monitor] channels inoperable.~~	~~D.1 Restore one [required hydrogen monitor] channel to OPERABLE status.~~	~~72 hours~~
~~E.~~ D. Required Action and associated Completion Time of Condition C ~~or D~~ not met.	~~E.1~~ D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
A4 ~~F.~~ E. As required by Required Action ~~E.1~~ D.1 and referenced in Table 3.3.3.1-1.	~~F.1~~ E.1 Be in MODE 3.	12 hours
~~G.~~ F. As required by Required Action ~~E.1~~ D.1 and referenced in Table 3.3.3.1-1.	~~G.1~~ F.1 Initiate action ~~in accordance with Specification 5.9.2.c.~~ P2 8 to prepare and submit a Special Report. 3.3.3.1 #17	Immediately

SURVEILLANCE REQUIREMENTS

P4

Applicable for

-------------------------------------NOTE-------------------------------------
~~These SRs apply to~~ each Function in Table 3.3.3.1-1.

SURVEILLANCE	FREQUENCY
SR 3.3.3.1.1 Perform CHANNEL CHECK.	31 days
P4 ------NOTE------ Applicable for each Function in Table 3.3.3.1-1 except Functions 10 and 11. SR 3.3.3.1.~~2~~ 3 Perform CHANNEL CALIBRATION. P4	[18] months
------NOTE------ Only applicable for Funtions 10 and 11 in Table 3.3.3.1-1 SR 3.3.3.1.2 Perform CHANNEL CALIBRATION.	92 days

3.3.2.1 #22

P4

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Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION ~~E~~ D.1
1. Reactor Steam Dome Pressure	2	~~F~~ E
2. Reactor Vessel Water Level - Wide Range	2	~~F~~ E
3. Reactor Vessel Water Level - Fuel Zone	2	E
4. ~~3.~~ Suppression Pool Water Level	2	~~F~~ E
6. ~~4.~~ Drywell Pressure	2	~~F~~ E
9. ~~5.~~ Primary Containment Area Radiation	2	~~10?~~ F
~~6. Drywell Sump Level~~	~~2~~	~~F~~
8. ~~7.~~ Drywell ~~Drain Sump Level~~ Area Radiation	2	F
12. ~~8.~~ PCIV Position Penetration Flow Path	2 per ~~valve~~ penetration flow path (b)(c) ~~(a)~~	~~F~~ E
~~9. Wide Range Neutron Flux~~	~~2~~	~~F~~
10. Drywell H_2 & O_2 Analyzer	2	~~F~~ E
11. Containment H_2 & O_2 Analyzer	2	~~F~~ E
7. ~~12.~~ Primary Containment Pressure	2	~~F~~ E
5. ~~13.~~ Suppression Pool Sector Water Temperature	2 ~~(b)~~ (a)	~~F~~ E

P3

3.3.3.1 #1,6,7

P4

P5

P3 ~~(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.~~

P5 (a) ~~(b)~~ Monitoring each ~~relief valve discharge location~~ of two sectors. B1

B1 ~~Reviewer's Note: Table 3.3.3.1-1 shall be amended for each plant as necessary to list:~~

~~1. All Regulatory Guide 1.97 Type A instruments, and~~

~~2. All Regulatory Guide 1.97 Category 1, non-Type A instruments specified in the plant's Regulatory Guide 1.97 Safety Evaluation Report.~~

(b) Not required for isolation valves whose associated penetration flow path is isolated.

(c) Only one position indication channel is required for penetration flow paths with only one control room indication channel.

3.3.3.1 #1,7

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level—Low Low, Level 2; and

b. Reactor Steam Dome Pressure—High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable. 3.3.4.2 #7	A.1	Restore channel to OPERABLE status.	14 days
	OR		
	A.2	---------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	14 days

(continued)

{Global Header change} and Drywell

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3.3.6.1 #25

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment and Drywell Isolation Instrumentation

LCO 3.3.6.1 The primary containment and drywell isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.b, 5.b, 5.d, and ~~5.c, 5.d,~~ ~~and~~ 5.e, and 5.f (P14) **AND** 3.3.6.1 #12 24 hours for Functions other than Functions 2.b, 5.b, 5.d, and ~~5.c, 5.d,~~ ~~and~~ 5.e, and 5.f (P14)
B. One or more automatic Functions with ~~primary containment~~ isolation capability not maintained.	B.1 Restore ~~primary containment~~ isolation capability. (C7)	1 hour

(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time of Condition F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	I.1 (Cb) Declare associated Standby Liquid Control ~~System~~ subsystem inoperable. OR I.2 Isolate the Reactor Water Cleanup System.	1 hour 1 hour
J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	J.1 Initiate action to restore channel to OPERABLE status. OR J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System suction from the reactor vessel. (3.3.6.1 #37)	Immediately Immediately

(continued)

INSERT 50A (3.3.6.1 #45)

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INSERT 50A

	OR	
J. (continued)	J.3.1 Initiate action to restore primary containment to OPERABLE status.	Immediately
	AND	
	J.3.2 ------NOTE------ Entry and exit is permissible under administrative control. ---------------- Initiate action to close one door in each primary containment air lock.	Immediately

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SURVEILLANCE REQUIREMENTS

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

3.3.6.1 #25

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each ~~Primary Containment Isolation~~ Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains ~~primary containment~~ isolation capability.

SURVEILLANCE		FREQUENCY	
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours	
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	[92] days	B1
SR 3.3.6.1.3	Calibrate the trip unit.	[92] days	B1
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	92 days	
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	[18] months	B1
SR 3.3.6.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	[18] months	B1
SR 3.3.6.1.7 (3.3.6.1 #38)	~~-------NOTE------- Radiation detectors may be excluded. ----------------~~ Verify the ISOLATION SYSTEM RESPONSE TIME for the Main Steam Isolation Valves is within limits.	[18] months on a STAGGERED TEST BASIS	B1

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3.3.6.1 #25

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

(Global Table change)

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	~~[~~2~~]~~ (B1)	D	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ (B1) SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~[-152.5]~~ -147 inches (B1)
b. Main Steam Line Pressure - Low	1	~~[~~2~~]~~ (B1)	E	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ (B1) SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ [837] psig (B1)
c. Main Steam Line Flow - High	1,2,3	~~[~~2~~]~~ per MSL (B1)	D	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ (B1) SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	~~≥ [176.5] psig~~ ≤ 151 psid, Line A ≤ 161 psid, Line B ≤ 158 psid, Line C ≤ 169 psid, Line D
d. Condenser Vacuum - Low	1,2(a), 3(a)	~~[~~2~~]~~ (B1)	D	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ (B1) SR 3.3.6.1.5 SR 3.3.6.1.6	≥ ~~[8.7]~~ 7.6 inches Hg vacuum (B1)
e. Main Steam Tunnel Temperature - High	1,2,3	~~[8]~~ 2 (B1)	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[194]~~ 148.5 °F (B1)
~~f. Main Steam Tunnel Differential Temperature - High~~	~~1,2,3~~	~~[2]~~ ~~(B1)~~	~~D~~	~~SR 3.3.6.1.1~~ ~~SR 3.3.6.1.2~~ ~~SR 3.3.6.1.5~~ ~~SR 3.3.6.1.6~~	~~≤ [104] °F~~ 61 ~~(B1)~~
~~g.~~ J. Manual Initiation	1,2,3	~~[~~2~~]~~	G	SR 3.3.6.1.6	NA (B1)
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	~~[~~2~~]~~(b) (B1)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[~~SR 3.3.6.1.3~~]~~ (B1) SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~	≥ ~~[-43.8]~~ -47 inches (B1)

3.3.6.1 #12

3.3.6.1 #25

3.3.6.1 #28

(continued)

(b) Also required to initiate the associated drywell isolation function.

(a) With any turbine [stop valve] not closed.

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
f. Main Steam Tunnel Area Temperature - High (El. 75 ft)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 145.3 °F
g. Main Steam Tunnel Area Temperature - High (El. 114 ft)	1,2,3	2	D		≤ 145.3 °F
h. Main Steam Line Turbine Shield Wall Temperature - High	1,2,3	2	D		≤ 111.3 °F
i. MSL Moisture Separator and Reheater Area Temperature - High	1,2,3	2	D		≤ 130 °F

P9

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Table 3.3.6.1-1 (page 2 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.3.6.1 #25 — 2. Primary Containment and Drywell Isolation (continued)		(b) 3.3.6.1 #25			
b. Drywell Pressure – High	1,2,3	~~[2]~~ (b) B1	K	SR 3.3.6.1.1 SR 3.3.6.1.2 B1 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ 3.3.6.1 #38	≤ ~~[1.44]~~ 1.88 psid B1
~~c. Reactor Vessel Water Level – Low Low Low, Level 1 (ECCS Divisions 1 and 2)~~	~~1,2,3~~	~~[2]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≥ [-152.5] inches~~
~~d. Drywell Pressure – High (ECCS Divisions 1 and 2)~~	~~1,2,3~~	~~[2]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [1.44] psig~~
B1 ~~e. Reactor Vessel Water Level – Low Low, Level 2 (HPCS)~~	~~1,2,3~~	~~[4]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≥ [-43.8] inches~~
~~f. Drywell Pressure – High (HPCS)~~	~~1,2,3~~	~~[4]~~	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [1.44] psig~~
P22 — c ~~g.~~ Containment ~~and Drywell Ventilation Exhaust~~ Purge Isolation Radiation – High	1,2,3	~~[2]~~ 1 B1	~~F~~ K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ e	≤ ~~[4.0] mR/hr~~ 1.57 R/hr B1
	~~[(b)]~~	~~[2]~~	~~K~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7~~	~~≤ [4.0] mR/hr~~

(continued)

3.3.6.1 #25

~~(b) During CORE ALTERATIONS, movement of irradiated fuel assemblies in [primary or secondary containment], or operations with a potential for draining the reactor vessel.~~

(b) Also required to initiate the associated drywell isolation function

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Table 3.3.6.1-1 (page 3 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.3.6.1 #25 — 2. Primary Containment and Drywell Isolation (continued)		3.3.6.1 #25			
d ~~k~~. Manual Initiation	1,2,3	[2] (b)	G	SR 3.3.6.1.6	NA (B1)
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	[1] (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] (B1) SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ 3.3.6.1 #38	≤ ~~[64]~~ 135.5 inches water (B1)
[b. RCIC Steam Line Flow Time Delay] (B1)	[1,2,3]	[1]	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ ~~[3]~~ seconds and ≤ ~~[7]~~ 13 seconds (B1)
c. RCIC Steam Supply Line Pressure – Low	1,2,3	[1] (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 (B1) [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ 3.3.6.1 #38	≥ ~~[53]~~ 55 psig (B1)
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	[2] (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 (B1) [SR 3.3.6.1.3] SR 3.3.6.1.5 SR 3.3.6.1.6	≤ [20] psig (B1)
e. RCIC Equipment Room Ambient Temperature – High	1,2,3	[1] (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[181]~~ 186.4 °F (B1)
3.3.6.1 #12 ~~f. RCIC Equipment Room Differential Temperature – High~~	~~1,2,3~~	~~[1]~~ (B1)	~~F~~	~~SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6~~	~~≤ [420] 99 °F~~ (B1)
f ~~g~~. Main Steam Line Tunnel Ambient Temperature – High	1,2,3	[1] (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[191]~~ 148.5 °F (B1)

(continued)

3.3.6.1 #25 (b) Also required to initiate the associated drywell isolation function

(LAR 93-14R1)

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
~~h. Main Steam Line Tunnel Differential Temperature - High~~	~~1,2,3~~	~~[1]~~ B1	~~F~~	~~SR 3.3.6.1.1~~ ~~SR 3.3.6.1.2~~ ~~SR 3.3.6.1.5~~ ~~SR 3.3.6.1.6~~	~~≤ [104]°F~~ 61 B1
g ~~i.~~ Main Steam Line Tunnel Temperature Timer	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	~~≤ [30] minutes~~ NA B1
h ~~j.~~ RHR Equipment Room Ambient Temperature - High	1,2,3	~~[1 per room]~~ 1 B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[171]~~ 121.1°F B1
~~k. RHR Equipment Room Differential Temperature - High~~	~~1,2,3~~	~~[1 per room]~~ 1 B1	~~F~~	~~SR 3.3.6.1.1~~ ~~SR 3.3.6.1.2~~ ~~SR 3.3.6.1.5~~ ~~SR 3.3.6.1.6~~	≤ ~~[102]~~ 33.6°F B1
i ~~l.~~ RCIC/RHR Steam Line Flow - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ B1 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[40]~~ 64.2 inches water B1
j ~~m.~~ Drywell Pressure - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ B1 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ P13	≤ ~~[1.44]~~ 1.89 psid B1
k ~~n.~~ Manual Initiation	1,2,3	~~[2]~~ 1	G	SR 3.3.6.1.6	NA E1
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ B1	≤ ~~[89]~~ 62.1 gpm B1
b. Differential Flow - Timer	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ ~~[57]~~ 47 seconds B1
c. RWCU Heat Exchanger Equipment Room Temperature - High	1,2,3	~~[1]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[126]~~ 107.5°F B1

3.3.6.1 #12

3.3.6.1 #3

(continued)

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Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

3.3.6.1 #13

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RWCU System Isolation (continued)					
d. RWCU Heat Exchanger Equipment Room Differential Temperature - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	42.5 ≤ ~~[66]~~ °F B1
d. ~~e.~~ RWCU Pump Rooms Temperature - High	1,2,3	~~[1]~~ 2 ~~[1 per room]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	169.5 ≤ ~~[176]~~ °F B1
f. RWCU Pump Rooms Differential Temperature - High	1,2,3	~~[1]~~ ~~[1 per room]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	82 ≤ ~~[40]~~ °F B1
e ~~g.~~ RWCU Valve Nest Room Temperature - High	1,2,3	~~[1]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	114.5 ≤ ~~[149]~~ °F B1
h. RWCU Valve Nest Room Differential Temperature - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	49.5 ≤ ~~[75]~~ °F B1
h ~~i.~~ Main Steam Line Tunnel Ambient Temperature - High (P11)	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	148.5 ≤ ~~[181]~~ °F B1
~~j.~~ Main Steam Line Tunnel Differential Temperature - High	1,2,3	~~[1]~~ B1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	61 ≤ ~~[104]~~ °F B1
L ~~k.~~ Reactor Vessel Water Level - Low Low, Level 2 (P14)	1,2,3	~~[2]~~ B1	F	B1 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ 3.3.6.1 #28	-47 ≥ ~~[-43.8]~~ inches B1
j ~~l.~~ Standby Liquid Control System Initiation	1,2	~~[1]~~ B1	I	SR 3.3.6.1.6	NA
~~m.~~ k Manual Initiation	1,2,3	~~[2]~~	G	SR 3.3.6.1.6	NA B1
RWCU Demineralizer Rooms Temperature - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 114.5°F
~~RWCU Demineralizer Rooms Differential Temperature - High~~	~~1,2,3~~	~~1~~	~~F~~	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	~~≤ 49.5°F~~
g RWCU Receiving Tank Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 114.5°F
~~RWCU Receiving Tank Room Differential Temperature - High~~	~~1,2,3~~	~~1~~	~~F~~	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	~~≤ 49.5°F~~

(continued)

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SR 3.3.6.1.1
SR 3.3.6.1.2
SR 3.3.6.1.5
SR 3.3.6.1.6

(LAR 93-14R1)

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment Isolation Instrumentation

(P8) FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ~~Shutdown Cooling~~ RHR System Isolation					
a. RHR Equipment Room Ambient Temperature – High	(C3) 1,2,3	~~[1 per room]~~ 2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[171]~~ 121.1 °F (B1)
(3.3.6.1 #12) ~~b. RHR Equipment Room Differential Temperature – High~~	~~2,3~~	~~[1 per room]~~	~~F~~	~~SR 3.3.6.1.1~~ ~~SR 3.3.6.1.2~~ ~~SR 3.3.6.1.5~~ ~~SR 3.3.6.1.6~~	~~≤ [102] °F~~ e
~~c.~~ b. Reactor Vessel Water Level – Low, Level 3	3,4,5 (d) (C3)	~~[2]~~ ~~(f)~~ (e) (P22) (B1)	J	SR 3.3.6.1.1 SR 3.3.6.1.2 (B1) ~~[~~SR 3.3.6.1.3~~]~~ e SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ e (3.3.6.1 #38)	≥ ~~[10.8]~~ 8.7 inches (B1)
	1,2,3 (c) (C3)	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ e (3.3.6.1 #38)	≥ 8.7 inches
c. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 ~~SR 3.3.6.1.7~~ e	≥ -147 inches (P14)
d. Reactor Steam Dome Pressure – High	1,2,3	~~[2]~~ e (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 (B1) ~~[~~SR 3.3.6.1.3~~]~~ e SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[~~150~~]~~ psig (B1)
e. Drywell Pressure – High	1,2,3	~~[2]~~ e (B1)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 (B1) ~~[~~SR 3.3.6.1.3~~]~~ e SR 3.3.6.1.5 SR 3.3.6.1.6	≤ ~~[1.43]~~ 1.88 psig e (B1)
f. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.6	NA (P3)

(P22) e ~~(f)~~ Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

(C3) (c) With reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure.
(d) With reactor steam dome pressure less than the RHR cut-in permissive pressure. (3.3.6.1 #21, 22)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.2.2	Perform CHANNEL FUNCTIONAL TEST.	[92] days B1
[SR 3.3.6.2.3	Calibrate the trip unit.	[92] days] B1
SR 3.3.6.2.4	Perform CHANNEL CALIBRATION.	[18] months B1
SR 3.3.6.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	[18] months B1
SR 3.3.6.2.6	------------------NOTE------------------ Radiation detectors may be excluded. -- Verify the ISOLATION SYSTEM RESPONSE TIME is within limits.	[18] months on a STAGGERED TEST BASIS B1

3.3.6.2 #35

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level — Low Low, Level 2 (B1)	1,2,3, ~~[(a)]~~	[2] (B1)	SR 3.3.6.2.1 SR 3.3.6.2.2 (B1) [SR 3.3.6.2.3] SR 3.3.6.2.4 SR 3.3.6.2.5 (B1) ~~[SR 3.3.6.2.6]~~	≥ ~~[-43.8]~~ -47 inches (B1)
2. Drywell Pressure — High	1,2,3	[2] (B1)	SR 3.3.6.2.1 SR 3.3.6.2.2 (B1) [SR 3.3.6.2.3] SR 3.3.6.2.4 SR 3.3.6.2.5 ~~[SR 3.3.6.2.6]~~ (B1)	≤ ~~[1.43]~~ 1.88 psig(d) (B1) 3.3.6.2 #35
(P5) ~~3. Fuel Handling Area Ventilation Exhaust Radiation — High High~~	~~1,2,3, [(a),(b)]~~	~~[2]~~	~~SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5 [SR 3.3.6.2.6]~~	~~≤ [4.0] mR/hr~~
~~4. Fuel Handling Area Pool Sweep Exhaust Radiation — High High~~	~~1,2,3, [(a),(b)]~~	~~[2]~~	~~SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5 [SR 3.3.6.2.6]~~	~~≤ [35] mR/hr~~ (P2)
[5. Manual Initiation	1,2,3, [(a)~~(b)~~]	~~[1 per group]~~ 2	SR 3.3.6.2.5	NA]

~~(a) During operations with a potential for draining the reactor vessel.~~ (P2)

(a) ~~(b)~~ During movement of irradiated fuel assemblies in the ~~[primary or secondary containment]~~ fuel building.

3. Fuel Building Ventilation Exhaust Radiation - High (1RMS * RE5A)	(a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.18 x 10^3 μCi/sec
4. Fuel Building Ventilation Exhaust Radiation - High (1RMS * RE5B)	(a)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 7.05 x 10^{-4} μCi/cc

Table 3.3.7.1-1 (page 1 of 1)
(B1) ~~[Control Room Fresh Air]~~ System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2 (B1)	1,2,3, ~~[(a)]~~	[2]	B	SR 3.3.7.1.1 SR 3.3.7.1.2 [SR 3.3.7.1.3] SR 3.3.7.1.4 SR 3.3.7.1.5	≥ ~~[-43.8]~~ -47 inches
2. Drywell Pressure - High (B1)	1,2,3	[2]	C	SR 3.3.7.1.1 SR 3.3.7.1.2 [SR 3.3.7.1.3] SR 3.3.7.1.4 SR 3.3.7.1.5	≤ ~~[1.43]~~ 1.38 psid
3. Control Room Ventilation Radiation Monitors	1,2,3, (a),(b)	~~[2]~~ 1 (B1)	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.5	≤ ~~[5] mR/hr~~ 0.97×10^{-5} µCi/cc (E1)

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during (C1) movement of irradiated fuel assemblies in the ~~[primary or secondary containment]~~. (B1)

3.3.7.1 #31

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

C1

3.3.1.1 #43

the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement. Functions not specifically credited in the accident analysis are retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, or a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis, but it is retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

3.3.1.1 #43

Six channels of Intermediate Range Monitor—Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux—High Function is required.

2.a. Average Power Range Monitor Neutron Flux—High, Setdown

The APRM channels receive input signals from the local power range monitors (LPRM) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor—Inop (continued)

APRM has too few LPRM inputs (< 11), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be inoperable without resulting in an RPS trip signal. This Function was not specifically credited in the accident analysis, but it is retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

3.3.1.1 #43

P9

Six ~~Four~~ channels of Average Power Range Monitor—Inop with three ~~two~~ channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

3. Reactor Vessel Steam Dome Pressure—High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure—High Function initiates a scram for transients that results in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 2, the reactor scram ~~terminates the MSIV closure event~~ (~~although~~ the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux—High signal, not the ~~Main Steam Isolation Valve—Closure~~ Reactor Vessel Steam Dome Pressure-High signal) ~~and~~, along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

C15

C8

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure—High Allowable Value is

(continued)

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Main Steam Isolation Valve—Closure (continued)

close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

7. Drywell Pressure—High

P9

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure—High Function is a secondary scram signal to Reactor Vessel Water Level—Low, Level 3 for large break LOCA events inside the drywell. This Function was not specifically credited in the accident analysis, but it is retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

3.3.1.1 #40

P9

High drywell pressure signals are initiated from four pressure transmitters that sense drywell to containment differential pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure—High Function, with two channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

8.a, b. Scram Discharge Volume Water Level—High

The SDV receives the water displaced by the motion of the CRD pistons during a reactor scram. Should this volume fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated when the remaining free volume is still sufficient to accommodate the water from a full core scram. However, even though the two types of Scram Discharge Volume Water Level—High Functions are an input to the RPS logic, no credit is taken for a scram initiated from these Functions for any of the design basis

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

3.3.1.1 #43

The reactor mode switch is a single switch with four channels, each of which inputs into one of the RPS logic channels.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

Four channels of Reactor Mode Switch—Shutdown Position Function, with two channels in each trip system, are available and required to be OPERABLE. The Reactor Mode—Switch Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

12. Manual Scram

The Manual Scram push button channels provide signals, via the manual scram logic channels, to each of the four RPS logic channels that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the ~~overall redundancy and diversity of the~~ RPS as required by the NRC approved licensing basis.

3.3.1.1 #43

There is one Manual Scram push button channel for each of the four RPS logic channels. In order to cause a scram it is necessary that at least one channel in each trip system be actuated.

(continued)

LAR 9[illegible]-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

normally

C19

CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift on one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

C19

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the ~~match~~ criteria, it may be an indication that the instrument has drifted outside its limit.

C19

The Frequency is based upon operating experience that demonstrates channel failure is rare. ~~The performance of the CHANNEL CHECK guarantees that undetected outright channel failure is limited to 12 hours.~~ The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

P9 ≥ 100%

3.3 #44

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate ~~within 2% RTP~~ ≥ 100% of calculated MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

restriction to ~~applying~~ satisfying this SR when < 25% RTP

C7

A ~~Note~~ is provided that only requires the SR to be met at ≥ 25% RTP because it is difficult to accurately ~~determine~~ Maintain APRM indication of core THERMAL POWER ~~from~~ consistent with a heat balance when < 25% RTP. At

(continued)

LAR 93-14R1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.10

The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days for SR 3.3.1.1.10 is based on the reliability analysis of Reference 9.

SR 3.3.1.1.11 ~~and~~, SR 3.3.1.1.13 and SR 3.3.1.1.17 (P6)

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations ~~. Measurement and setpoint error historical determinations must be performed~~ consistent with the plant specific setpoint methodology. ~~The channel shall be left calibrated consistent with the assumptions of the setpoint methodology.~~ (C20)

~~If the as found setpoint is not within its required Allowable Value, the plant specific setpoint methodology may be revised, as appropriate, if the history and all other pertinent information indicate a need for the revision. The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

3.3.1.1 #45

Note 1 states that

~~As noted,~~ neutron detectors and flow reference transmitters (P6) are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8). The Frequency of SR 3.3.1.1.11 , SR 3.3.1.1.13, (C21) and SR 3.3.1.1.17 (P6) is based upon the assumption ~~of a 184 day calibration interval in the determination~~ (C7) INSERT B29A of the magnitude

(P6) Calibration of the flow reference transmitters is performed on an 18 month Frequency (SR 3.3.1.1.17).

(continued)

(LAK 93-14X1)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.1 and SR 3.3.1.2.3 (continued)

more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.2.2

C6 containing fuel.

To provide adequate coverage of potential reactivity changes in the core, one SRM is required to be OPERABLE in the quadrant where CORE ALTERATIONS are being performed, and the other OPERABLE SRM must be in an adjacent quadrant. Note 1 states that this SR is required to be met only during CORE ALTERATIONS. It is not required to be met at other times in MODE 5 since core reactivity changes are not occurring. This Surveillance consists of a review of plant logs to ensure that SRMs required to be OPERABLE for given CORE ALTERATIONS are, in fact, OPERABLE. In the event that only one SRM is required to be OPERABLE, per Table 3.3.1.2-1, footnote (b), only the a. portion of this SR is required. Note 2 clarifies that the three requirements can be met by the same ~~or different~~ OPERABLE SRMs. The 12 hour Frequency is based upon operating experience and supplements operational controls over refueling activities, which include steps to ensure that the SRMs required by the LCO are in the proper quadrant.

C1 more than one of

3.3.1.2 # 11

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate. This ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. Verification of the signal to noise ratio also ensures that the detectors are inserted to a normal operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while fully withdrawn is assumed to be "noise" only. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.4 (continued)

of irradiated fuel assemblies, to establish the minimum count rate.

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated quadrant, even with a control rod withdrawn the configuration will not be critical.

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is relaxed from 12 hours to 24 hours.

3.3.1.2 #10

SR 3.3.1.2.5 ~~and SR 3.3.1.2.6~~

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. ~~SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This 7 day Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.~~

~~SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below and in MODES 3 and 4. Since core reactivity changes do not normally take place, the Frequency has been extended from 7 days to 31 days.~~ The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

The Note to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 31 day

(continued)

BASES

3.3.1.2 #10

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 ~~and SR 3.3.1.2.6~~ (continued)

Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

SR 3.3.1.2.~~7~~ 6 3.3.1.2 #10.

Performance of a CHANNEL CALIBRATION verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range, and with an accuracy specified for a fixed useful life.

C2

INSERT B42A

REFERENCES None.

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BASES

3.3.2.1 #17

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Rod Pattern Controller (continued)

Control

compliance with BPWS are specified in LCO 3.1.6, "Rod Pattern ~~Control~~."

to satisfy Criterion 3 C8

The Rod Pattern Controller Function satisfies Criterion 3 of the NRC Policy Statement. Since the RPC is a backup to operator control of control rod sequences, only a single channel is required OPERABLE (Ref. 6). However, the RPC is designed as a dual channel system and will not function without two OPERABLE channels. Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing individual control rods in the Rod Action Control System (RACS) to allow continued operation with inoperable control rods or to allow correction of a control rod pattern not in compliance with the BPWS. The individual control rods may be bypassed as required by the conditions, and the RPC is not considered inoperable provided SR 3.3.2.1.~~8~~ 9 is met.

C8 would be C2 C4

Compliance with the BPWS, and therefore OPERABILITY of the RPC, is required in MODES 1 and 2 with THERMAL POWER ≤ ~~10%~~ RTP. When THERMAL POWER is > ~~10%~~ RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA. In MODES 3 and 4, all control rods are required to be inserted in the core. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

20 20

2. Reactor Mode Switch—Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch—Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch—Shutdown Position Function satisfies Criterion 3 of the NRC Policy Statement.

(continued)

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BASES

ACTIONS

B.1 (continued)

inoperable if individual control rods are bypassed in the RACS as required by LCO 3.1.3 or LCO 3.1.6. Under these conditions, continued operation is allowed if the bypassing of control rods and movement of control rods is verified by a second licensed operator or other qualified member of the technical staff per SR 3.3.2.1.~~8~~ 9. C4

C.1 and C.2

3.3.2.1 #16

If one Reactor Mode Switch—Shutdown Position control rod withdrawal block channel is inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. Required Action C.1 and Required Action C.2 are consistent with the normal action of an OPERABLE Reactor Mode Switch—Shutdown Position Function to maintain all control rods inserted. Therefore, there is no distinction between Required Actions for the Conditions of one or two channels inoperable. In both cases (one or both channels inoperable), suspending all control rod withdrawal immediately, and immediately initiating action to fully inserting all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical, with adequate SDM ensured by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

G1

[Reviewer's Note: Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.]

As noted at the beginning of the SR, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are also modified by a Note to indicate that when an RWL channel is placed in an inoperable status solely for performance of required Surveillances, entry into

C6

(continued)

Individual control rods may also be required to be bypassed to allow continuous withdrawl for determining the location of leaking fuel assemblies, adjustment of control rod speed, or control rod scram time testing.

BASES

C4

SURVEILLANCE REQUIREMENTS

3.3.2.1 #15

SR 3.3.2.1.~~8~~ 9 (continued)

control rod or correction of a control rod pattern not in compliance with BPWS. With the control rods bypassed in the RACS, the RPC will not control the movement of these bypassed control rods. To ensure the proper bypassing and movement of those affected control rods, a second licensed operator or other qualified member of the technical staff must verify the bypassing and movement of these control rods is in conformance with applicable analyses. Compliance with this SR allows the RPC and RWL to be OPERABLE with these control rods bypassed.

REFERENCES

P1 B1

1. UFSAR, Section ~~[~~7.6.1.7.~~3~~e~~]~~.
2. UFSAR, Section ~~[~~15.4.2~~]~~.
3. NEDE-24011-P-A~~-9-US~~, "General Electrical Standard Application for Reload Fuel," ~~Supplement for United States, Section S 2.2.3.1, September 1988.~~ (latest approved revision). P10
4. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners Group, July 1986.
5. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
6. NRC SER, Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.
7. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine whether a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to obtain an estimate of the magnitude of any impending threat.

The plant specific Regulatory Guide 1.97 analysis (Ref. 2) documents the process that identified Type A and Category I, non-Type A, variables.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category I, non-Type A, instrumentation is retained in the Technical Specifications (TS) because it is intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I, non-Type A, variables are important for reducing public risk.

P5

LCO

LCO 3.3.3.1 requires at least two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident.

Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. ~~[More than two channels may be required at some units if the Regulatory Guide 1.97 analysis determined that failure of one accident monitoring channel results in information ambiguity (e.g., the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]~~ e

B1

3.3.3.1 #1

The exception ~~of~~ to the two channel requirement is primary containment isolation valve (PCIV) position. In this case, the important information is the status of the primary containment penetrations. The LCO requires ~~one~~ two position indicators for each ~~active PCIV~~ penetration flow path. This is sufficient to

(continued)

BASES

LCO (continued)

3.3.3.1 #12

redundantly verify the isolation status of each isolable penetration either via indicated status of the ~~active~~ automatic valve(s) and prior knowledge of passive valve or via system boundary status. If a normally ~~active~~ automatic PCIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. INSERT B54B

3.3.3.1 #1,7

Listed below is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1, in the accompanying LCO. ~~These discussions are intended as examples of what should be provided for each Function when the plant specific Bases are prepared.~~ B1

1. Reactor Steam Dome Pressure

Reactor steam dome pressure is a Category I variable provided to support monitoring of Reactor Coolant System (RCS) integrity and to verify operation of the Emergency Core Cooling Systems (ECCS). Two independent pressure transmitters with a range of 0 psig to 1500 psig monitor pressure. Wide range recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

P3 2,3. Reactor Vessel Water Level and Fuel Zone

Reactor vessel water level is a Category I variable provided to support monitoring of core cooling and to verify operation of the ECCS. The wide range water level channels provide the PAM Reactor Vessel Water Level Function. The wide range water level channels ~~measure from 17 inches below the dryer skirt down to a point just below the bottom of the active fuel.~~ e Wide range water level is measured by two independent differential pressure transmitters. The output from these channels is recorded on two independent pen recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

P3 INSERT B54A

(continued)

INSERT B54A

two inches above the top of active fuel to 218 inches above the top of active fuel. The fuel zone water level channels overlap with the wide range channels and measure down to the bottom of the active fuel. Both the wide range and the fuel zone water levels are

INSERT B54B

In addition, Note (c) of Table 3.3.3.1-1 requires only one position indication for those penetrations which only have one position indication provided to the control room.

3.3.3.1
#1

BASES

LCO

9 ~~5.~~ Primary Containment Area Radiation (High Range) (continued)

operators in determining the need to invoke site emergency plans.

B1 [~~For this plant~~, primary containment area radiation (high range) PAM instrumentation consists of ~~the following:~~] e INSERT B56A P3

P3

~~6. Drywell Sump Level~~

~~Drywell sump level is a Category I variable provided for verification of ECCS functions that operate to maintain RCS integrity.~~

~~[For this plant, the drywell sump level PAM instrumentation consists of the following:]~~

~~7. Drywell Drain Sump Level~~

~~Drywell drain sump level is a Category I variable provided to detect breach of the RCPB and for verification and long term surveillance of ECCS functions that operate to maintain RCS integrity.~~

~~[For this plant, the drywell drain sump level PAM instrumentation consists of the following:]~~

3.3.3.1 #1, 12

12 ~~8.~~ Penetration Flow Path Primary Containment Isolation Valve (PCIV) Position

PCIV position is provided for verification of containment integrity. In the case of PCIV position, the important information is the status of the containment penetration flow path. ~~The LCO requires one position indicator for each active PCIV.~~ e INSERT B56B This is sufficient to verify redundantly the isolation status of each isolable penetration via indicated status of the ~~active~~ automatic valve and, as applicable, prior knowledge of passive valve or system boundary status. If a penetration is isolated, position indication for ~~both of~~ the PCIV~~S~~ in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration is not required to be OPERABLE. INSERT B56C

(continued)

LAR 93-14R1

INSERT B56A

two high range containment area radiation signals transmitted from separate radiation elements and continuously recorded and displayed on two control room recorders. The recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

8. Drywell Area Radiation (High Range)

Drywell area radiation (high range) is a Category I variable provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Drywell area radiation (high range) PAM instrumentation consists of two high range drywell area radiation signals transmitted from separate radiation elements and continuously recorded and displayed on two control room recorders. The recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

INSERT B56B

The LCO requires one channel of valve position indication in the control room to be OPERABLE for each automatic PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with two automatic valves. For containment penetrations with only one automatic PCIV having control room indication, Note (c) ~~(b)~~ requires a single channel of valve position indication to be OPERABLE.

INSERT B56C

by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured,

3.3.3.1 #1

LAR 93-1421

BASES

Penetration Flow Path,

LCO

3.3.3.1 #1

12 ~~8.~~ Primary Containment Isolation Valve (PCIV) Position (continued)

[~~For this plant,~~ the PCIV position PAM instrumentation consists of ~~the following:~~ INSERT B57B]

P3

9. Wide Range Neutron Flux

Wide range neutron flux is a Category I variable provided to verify reactor shutdown.

[For this plant, wide range neutron flux PAM instrumentation consists of the following:]

10, 11. Drywell and Containment Hydrogen ~~and Oxygen~~ Analyzer

P3

Drywell and containment hydrogen ~~and oxygen~~ analyzers are Category I instruments provided to detect high hydrogen ~~or oxygen~~ concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions.

P3

B1

[~~For this plant,~~ the drywell and containment hydrogen ~~and oxygen~~ analyzers PAM instrumentation consists of ~~the following:~~ INSERT B57A]

P3

7 ~~12.~~ Primary Containment Pressure

Primary containment pressure is a Category I variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two wide range primary containment pressure signals are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders. These recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

(continued)

INSERT B57A

containment and drywell hydrogen concentration signals transmitted from two separate hydrogen analyzers and recorded on two two-pen recorders in the control room. One pen records the hydrogen concentration and one pen records the sample point on each of the two independent recorders. Measurement capability is provided over the range of 0 to 10 percent hydrogen concentration using a sample drawing system.

INSERT B57B

individual position indication (open-closed) in the control room for each automatic containment isolation valve as described in USAR Section 7.5 (Reference 3).

3.3.3.1
#1

ACTIONS (continued)

function of the instruments, the operator's ability to diagnose an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

C4

divisions

A Note has also been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent ~~trains~~, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate inoperable functions. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel(s) (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

P5

3.3.3.1 #17

to prepare and submit a special report

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of actions ~~in accordance with Specification 5.9.2.c, "Special Reports," which requires a written report, approved by the [onsite review committee],~~ to be ~~submitted~~ to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This Action is appropriate in lieu of a shutdown

C2

5.0 #131

The Special Report shall be submitted in accordance with 10 CFR 50.4 within 14 days of entering Condition B.

(continued)

BASES

ACTIONS (P4) ~~E.1~~ D.1 (continued)

Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C ~~or D, as applicable,~~ and the associated Completion Time has expired, Condition ~~E~~ D is entered for that channel and provides for transfer to the appropriate subsequent Condition. (P4)

(P4) ~~F.1~~ E.1

For the majority of Functions in Table 3.3.3.1-1, if any Required Action and associated Completion Time of Condition C ~~or D~~ is not met, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

(P4) ~~G.1~~ F.1

3.3.3.1 #17 (C3) Since alternate means of monitoring ~~reactor vessel water level and~~ primary containment area radiation have been developed and tested, the Required Action is not to shut down the plant but rather to ~~follow the directions of Specification 5.9.2.c.~~ initiate action to prepare and submit a special report to the NRC. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. The special report shall be submitted in accordance with 10 CFR 50.4 within 14 days of entering Condition F. 5.0 #131

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation Function in Table 3.3.3.1-1, except as noted 3.3.3.1 #19

(continued)

LAR 93-14R1

BASES (continued)

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," ~~[Date]~~ Revision 2, December 1980.

B1

2. ~~[Plant specific documents (e.g., FSAR, NRC Regulatory Guide 1.97, SER letter).]~~ NRC Safety Evaluation Report, "Conformance to Regulatory Guide 1.97, Revision 2, River Bend Station, Unit 1," dated June 30, 1986.

3.3.3.1 #1

3. USAR Section 7.5.

LAR 95-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.1 (continued)

verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

C4

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized. ~~Performance of a CHANNEL CHECK guarantees that undetected outright channel failure is limited to 31 days.~~

3.3.3.2
#7

The Frequency is based upon plant operating experience that demonstrates channel failure is rare.

SR 3.3.3.2.2

Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check.

SR 3.3.3.2.2 verifies each required Remote Shutdown System transfer switch and control circuit performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. However, this Surveillance is not required to be performed only during a plant outage. Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

P3

Valve position Functions are excluded since channel performance is adequately determined during performance of other valve surveillances.

The 18 month Frequency is based upon operating experience and is consistent with the typical industry refueling cycle.

(continued)

BASES (continued)

REFERENCES

p2

1. 10 CFR 50, Appendix A, GDC 19.
2. RBS Technical Requirements Manual

3.3.3.2
#8

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

TCV Fast Closure, Trip Oil Pressure—Low (continued)

Closure, Trip Oil Pressure—Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

3.3.4.1 #18

Fast closure of the TCVs is determined by measuring the EHC fluid pressure at each control valve. There is one pressure transmitter associated with each control valve, and the signal from each transmitter is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure—Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 40% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 40% RTP. Four channels of TCV Fast Closure, Trip Oil Pressure—Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the analysis, whenever the THERMAL POWER is ≥ 40% RTP with any recirculating pump in fast speed. Below 40% RTP or with recirculation pumps in slow speed, the Reactor Vessel Steam Dome Pressure—High and the APRM Fixed Neutron Flux—High Functions of the RPS are adequate to maintain the necessary safety margins. The turbine first stage pressure/reactor power relationship for the setpoint of the automatic enable is identical to that described for TSV closure.

ACTIONS

(E1)

~~[Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.]~~

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.6 (continued)

accident analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criteria are included in Reference 6.

A Note to the Surveillance states that breaker interruption time may be assumed from the most recent performance of SR 3.3.4.1.7. This is allowed since the time to open the contacts after energization of the trip coil and the arc suppression time are short and do not appreciably change, due to the design of the breaker opening device and the fact that the breaker is not routinely cycled.

3.3.4.1 #8

Each test shall include at least the logic of one type of channel input (Turbine Stop Valve Closure or Turbine Control Valve Fast Closure, Trip Oil Pressure - Low) such that both types of channel inputs are [illegible] at least once per 36 months.

EOC-RPT SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Response times cannot be determined at power because operation of final actuated devices is required. Therefore, ~~the 18 month~~ this Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components that cause serious response time degradation, but not channel failure, are infrequent occurrences. C5

SR 3.3.4.1.7

P8

This SR ensures that the RPT breaker interruption time ~~(arc suppression time plus time to open the contacts)~~ is provided to the EOC-RPT SYSTEM RESPONSE TIME test. The 60 month Frequency of the testing is based on the difficulty of performing the test and the reliability of the circuit breakers.

INSERT B30A

REFERENCES

1. UFSAR, ~~Figure [ ] (EOC-RPT instrumentation logic)~~ Section 7.6.1.1.
2. UFSAR, Section 5.2.2.
3. UFSAR, Sections 15.1.1, 15.1.2, and 15.1.3.
4. UFSAR, Sections ~~[5.5.16.1]~~ 15.2.2, 15.2.3 and ~~[7.6.10]~~ 15.2.5.

P1 B1

(continued)

LAR 93-14R1

ACTIONS (continued)

of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ATWS-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable ATWS-RPT instrumentation channel.

<change deleted>

3.3.4.2 #7

A.1 and A.2

With one or more channels inoperable, but with ATWS-RPT capability for each Function maintained (refer to Required Action B.1 and C.1 Bases), the ATWS-RPT System is capable of performing the intended function. However, the reliability and redundancy of the ATWS-RPT instrumentation is reduced, such that a single failure in the remaining trip system could result in the inability of the ATWS-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desirable to place the channel in trip (e.g., as in the case where placing the inoperable channel would result in an P3 ATWS- RPT), or if the inoperable channel is the result of an inoperable breaker, Condition D must be entered and its Required Actions taken.

(continued)

LAR 93-14R1

ACTIONS (continued)

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when sufficient channels are OPERABLE or in trip such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal, and both recirculation pumps can be tripped. This requires two channels of the Function in the same trip system to each be OPERABLE or in trip, and the four motor breakers (two fast speed and two LFMG) to be OPERABLE or in trip.

3.3.4.2 #7

P2 insert deleted

The 72 hour Completion Time is sufficient for the operator to take corrective action (e.g., restoration or tripping of channels) and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period and the fact that one Function is still maintaining ATWS-RPT trip capability.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within both Functions result in both Functions not maintaining ATWS-RPT trip capability. The description of a Function maintaining ATWS-RPT trip capability is discussed in the Bases for Required Action B.1, above.

The 1 hour Completion Time is sufficient for the operator to take corrective action and takes into account the likelihood of an event requiring actuation of the ATWS-RPT instrumentation during this period.

D.1 and D.2

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours (Required Action D.2). Alternately, the associated recirculation pump may be removed from service since this

(continued)

BACKGROUND

Low Pressure Coolant Injection Subsystems (continued)

minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the analyses.

(P5) The RHR test line (suppression pool cooling) isolation ~~and suppression pool spray isolation~~ valves (which are also PCIVs) are closed on a LPCI initiation signal to allow full system flow assumed in the accident analysis and maintain containment isolated in the event LPCI is not operating.

3.3.5.1 #29

The LPCI subsystems monitor the pressure in the reactor vessel to ensure that, prior to an injection valve opening, the reactor pressure has fallen to a value below the LPCI subsystem's maximum design pressure. The variable is monitored by four (two) redundant transmitters per Division, which are, in turn, connected to four trip units. The outputs of the four Division 2 LPCI (loops B and C) trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic. The Division 1 LPCI (loop A) receives its signal from the LPCS logic, which uses a similar one-out-of-two taken twice logic.

High Pressure Core Spray System

The HPCS System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low, Level 2 or Drywell Pressure—High. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each variable. The HPCS System initiation signal is a sealed in signal and must be manually reset.

The HPCS pump discharge flow is monitored by a flow transmitter. When the pump is running and discharge flow is low enough that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow full system flow assumed in the accident analyses.

The HPCS test line isolation valve (which is also a PCIV) is closed on a HPCS initiation signal to allow full system flow

(continued)

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.~~e~~f, 1.~~f~~g, 2.~~e~~f. Low Pressure Coolant Injection and Low Pressure Core Spray Pump Discharge Flow—Low (Bypass) (continued)

LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.~~g~~h, 2.~~f~~g. Manual Initiation

The Manual Initiation push button channels introduce signals into the appropriate ECCS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There is one push button for each of the two Divisions of low pressure ECCS (i.e., Division 1 ECCS, LPCS and LPCI A; Division 2 ECCS, LPCI B and LPCI C).

3.3.5.1 #21

The Manual Initiation Function is not assumed in any accident or transient analyses in the UFSAR. However, the Function is retained for ~~overall redundancy and diversity of~~ the low pressure ECCS function as required by the NRC in the plant licensing basis. P1

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons. Each channel of the Manual Initiation Function (one channel per Division) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

High Pressure Core Spray System

3.a. Reactor Vessel Water Level—Low Low, Level 2

C4

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCS System and associated DG ~~is~~ are initiated at Level 2 to maintain level above the top of the active fuel. The Reactor Vessel Water Level—Low Low, Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCS during the transients analyzed in References 1 and 3. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with HPCS is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling

(continued)

LAR 93-14P1

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.f, 3.g. HPCS Pump Discharge Pressure—High (Bypass) and HPCS System Flow Rate—Low (Bypass) (continued)

of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow transmitter is used to detect the HPCS System's flow rate. The logic is arranged such that the transmitter causes the minimum flow valve to open, provided the HPCS pump discharge pressure, sensed by another transmitter, is high enough (indicating the pump is operating). The logic will close the minimum flow valve once the closure setpoint is exceeded. (The valve will also close upon HPCS pump discharge pressure decreasing below the setpoint.)

The HPCS System Flow Rate—Low and HPCS Pump Discharge Pressure—High Allowable Value is high enough to ensure that pump flow rate is sufficient to protect the pump, yet low enough to ensure that the closure of the minimum flow valve is initiated to allow full flow into the core. The HPCS Pump Discharge Pressure—High Allowable Value is set high enough to ensure that the valve will not be open when the pump is not operating.

One channel of each Function is required to be OPERABLE when the HPCS is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for HPCS Applicability Bases.

3.h. Manual Initiation

The Manual Initiation push button channel introduces a signal into the HPCS logic to provide manual initiation capability and is redundant to the automatic protective instrumentation. There is one push button for the HPCS System.

3.3.5.1 #31

The Manual Initiation Function is not assumed in any accident or transient analysis in the UFSAR. However, the Function is retained for ~~overall redundancy and diversity of~~ the HPCS function as required by the NRC in the plant licensing basis. P1

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of the Manual

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.g, 5.f. ADS Bypass Timer (High Drywell Pressure) (continued)

is chosen to be short enough ~~that~~ (C4) so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Four channels of the ADS Bypass Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.h, 5.g. Manual Initiation

The Manual Initiation push button channels introduce signals into the ADS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There are two push buttons for each ADS trip system (total of four).

3.3.5.1 #21

The Manual Initiation Function is not ~~assumed~~ in any accident or transient analyses in the U~~F~~SAR. However, the Function is retained for ~~overall redundancy and diversity of~~ (P1) the ADS function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push buttons. Four channels of the Manual Initiation Function (two channels per ADS trip system) are only required to be OPERABLE when the ADS is required to be OPERABLE. Refer to LCO 3.5.1 for ADS Applicability Bases.

ACTIONS

(B1)

~~Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent ~~trains~~, subsystems, components, or variables

(C5) divisions

(continued)

LAR 93-14R1

BASES

ACTIONS

3.3.5.1 #17

INSERT B13-4

F.1 and F.2 (continued)

channel to OPERABLE status if both HPCS and RCIC are OPERABLE. If either HPCS or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action F.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

G.1 and G.2

Required Action G.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within similar ADS trip system Functions result in automatic initiation capability being lost for the ADS. Automatic initiation capability is lost if either (a) one Function 4.c channel and one Function 5.c channel are inoperable, (b) one or more Function 4.e channels and one or more Function 5.e channels are inoperable, (c) one or more Function 4.f channels and one or more Function 5.e channels are inoperable, or (d) one or more Function 4.g channels and one or more Function 5.f channels are inoperable.

In this situation (loss of automatic initiation capability), the 96 hour or 3 day allowance, as applicable, of Required Action G.2 is not appropriate, and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability in both trip systems. The Note to Required Action G.1 states that Required Action G.1 is only applicable for Functions 4.c, 4.e, 4.f, 4.g, 5.c, 5.e,

(continued)

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INSERT B123A

Since RCIC is not required to be OPERABLE at reactor steam dome pressures < 150 psig, RCIC will not be considered inoperable below this pressure regardless of the actual status or capability of RCIC.

3.3.5.1 #17

LAR 93-14(R1)

ACTIONS

G.1 and G.2 (continued)

and 5.f. Required Action G.1 is not applicable to Functions 4.h and 5.g (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 96 hours or 8 days (as allowed by Required Action G.2) is allowed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action G.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable channels within similar ADS trip system Functions, as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

3.3.5.1 #17

INSERT B124A Deleted

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status if both HPCS and RCIC are OPERABLE (Required Action G.2). If either HPCS or RCIC is inoperable, the time is reduced to 96 hours. If the status of HCPS or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCS or RCIC inoperability. However, total time for an inoperable channel cannot exceed 8 days. If the status of HPCS or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would not necessarily result in a safe state for the channel in all events.

(continued)

LAR 93-14R1

INSERT B124A

Since RCIC is not required to be OPERABLE at reactor steam dome pressures < 150 psig, RCIC will not be considered inoperable below this pressure regardless of the actual status or capability of RCIC.

3.3.5.1 #17

LAR 93-14(1)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Manual Initiation

The Manual Initiation push button switch introduces a signal into the RCIC System initiation logic that is redundant to the automatic protective instrumentation and provides manual initiation capability. There is one push button for the RCIC System.

3.3.5.2 #13

The Manual Initiation Function is not assumed in any accident or transient analyses in the U FSAR. However, the Function is retained for ~~overall redundancy and diversity of~~ the RCIC function as required by the NRC in the plant licensing basis. P1

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of Manual Initiation is required to be OPERABLE when RCIC is required to be OPERABLE.

ACTIONS

B1

~~[Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the NRC staff Safety Evaluation Report (SER) for the topical report.]~~

A Note has been provided to modify the ACTIONS related to RCIC System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent ~~trains~~ divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RCIC System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RCIC System instrumentation channel.

C1

(continued)

<Global Header change> and Drywell LAR 93-14r1

3.3.6.1 #25

and Drywell

B 3.3 INSTRUMENTATION

B 3.3.6.1 Primary Containment Isolation Instrumentation

BASES

3.3.6.1 #25

and drywell — and drywell isolation valves

BACKGROUND

The primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs). The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

INSERT B 141A

3.3.6.1 #25

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary containment and reactor coolant pressure boundary (RCPB) isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a primary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) ambient ~~and differential~~ temperatures, (c) main steam line (MSL) flow measurement, (d) Standby Liquid Control (SLC) System initiation, (e) condenser vacuum loss, (f) main steam line pressure, (g) reactor core isolation cooling (RCIC) and RCIC/residual heat removal (RHR) steam line flow, (h) ventilation exhaust radiation, (i) RCIC steam line pressure, (j) RCIC turbine exhaust diaphragm pressure, (k) reactor water cleanup (RWCU) differential flow, (l) reactor steam dome pressure, and (m) drywell pressure. Redundant sensor input signals are provided from each such isolation initiation parameter. The only exception is SLC System initiation. In addition, manual isolation of the logics is provided.

3.3.6.1 #13

3.3.6.1 #25 — and drywell

The primary containment isolation instrumentation has inputs to the trip logic from the isolation Functions listed below.

(continued)

INSERT B141A

The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the the primary containment.

3.3.6.1
#25

LAR 93-14(1)

BASES

BACKGROUND (continued)

1. Main Steam Line Isolation

Most Main Steam Line Isolation Functions receive inputs from four channels. The outputs from these channels are combined in one-out-of-two taken twice logic to initiate isolation of all main steam isolation valves (MSIVs). The outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate all MSL drain valves. Each MSL drain line has two isolation valves with one two-out-of-two logic system associated with each valve.

The exception to this arrangement is the Main Steam Line Flow—High Function. This Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of four trip strings. Two trip strings make up each trip system, and both trip systems must trip to cause an MSL isolation. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings within a trip system are arranged in a one-out-of-two taken twice logic. Therefore, this is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation of the MSIVs. Similarly, the 16 flow channels are connected into two two-out-of-two logic trip systems (effectively, two one-out-of-four twice logic), with each trip system isolating one of the two MSL drain valves.

3.3.6.1 #25

2. Primary Containment and Drywell Isolation

3.3.6.1 #25

Each Primary Containment and Drywell Isolation Function receives inputs from four channels. The outputs from these channels are arranged into two two-out-of-two logic trip systems. One trip system initiates isolation of all inboard PCIVs and drywell isolation valves, while the other trip system initiates isolation of all outboard PCIVs and drywell isolation valves. Each trip system logic closes one of the two valves on each penetration so that operation of either trip system isolates the penetration.

3. Reactor Core Isolation Cooling System Isolation

3.3.6.1 #12

Most Functions receive input from two channels, with each channel in one trip system using one-out-of-one logic. Function~~s~~ 3.j ~~and 3.k~~ (RHR Equipment Room Temperature) ~~have~~ has one channel in each trip system in each room for a total of four channels ~~per Function~~, but the logic is the same

(continued)

(LAR 93-14R1)

BASES

BACKGROUND

3. Reactor Core Isolation Cooling System Isolation (continued)

(one-out-of-one). Each of the two trip systems is connected to one of the two valves on each RCIC penetration so that operation of either trip system isolates the penetration. The exception to this arrangement is the RCIC Turbine Exhaust Diaphragm Pressure—High Function. This Function receives input from four turbine exhaust diaphragm pressure channels. The outputs from the turbine exhaust diaphragm pressure channels are connected into two two-out-of-two trip systems, each trip system isolating one of the two RCIC valves. There is one manual isolation switch which can isolate only the outboard RCIC System containment isolation valves. P5

4. Reactor Water Cleanup System Isolation

3.3.6.1 #12

Most Functions receive input from two channels with each channel in one trip system using one-out-of-one logic. Functions 4.~~g~~d and 4.~~f~~e (RWCU Pump Room Temperature) ~~have~~ has one channel in each trip system in each room for a total of four channels ~~per Function~~, but the logic is the same (one-out-of-one). Each of the two trip systems is connected to one of the two valves on each RWCU penetration so that operation of either trip system isolates the penetration. The exception to this arrangement is the Reactor Vessel Water Level—Low Low, Level 2 Function. This Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into two two-out-of-two trip systems, each trip system isolating one of the two RWCU valves.

P8 RHR

RHR System

5. ~~Shutdown Cooling~~ System Isolation

P14 Reactor Vessel Water Level-Low Low Low, Level 1;

The ~~Shutdown Cooling~~ Isolation Function receives input signals from instrumentation for the Reactor Vessel Water Level—Low, Level 3; Drywell Pressure—High; Reactor Steam Dome Pressure—High; and RHR Equipment Room Ambient ~~and Differential~~ Temperature—High Functions. The Reactor Vessel Water Level—Low, Reactor Steam Dome Pressure—High, and Drywell Pressure—High Functions each have four channels. The outputs from the reactor vessel water level and drywell pressure channels are connected into two two-out-of-two trip systems. The reactor steam dome pressure is arranged into two one-out-of-two trip systems.

3.3.6.1 #12

P14 Reactor Vessel Water Level - Low Low Low,

(continued)

LAR 93-14R1

BASES

P8A

BACKGROUND

RHR

5. ~~Shutdown Cooling~~ System Isolation (continued)

3.3.6.1 #12

The RHR Equipment Room Ambient ~~and Differential~~ Temperature Functions receive input from four channels with each channel in one trip system in one room using one-out-of-one logic. Each of the two trip systems is connected to one of the two valves on each shutdown cooling penetration so that operation of either trip system isolates the penetration.

3.3.6.1 #25

and drywell

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases, for more detail.

INSERT B144A

Primary containment isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement. Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

and drywell

The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time, where appropriate.

3.3.6.1 #25

and Drywell

Allowable Values are specified for each Primary Containment Isolation Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and

(continued)

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INSERT B144A

The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the the primary containment. Refer to LCO 3.6.5.3, "Drywell Isolation Valves," Applicable Safety Analysis Bases, for more detail.

3.3.6.1
#25

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

Certain Emergency Core Cooling Systems (ECCS) and RCIC valves (e.g., minimum flow) also serve the dual function of automatic PCIVs. The signals that isolate these valves are also associated with the automatic initiation of the ECCS and RCIC. The instrumentation and ACTIONS associated with these signals are addressed in LCO 3.3.5.1, "ECCS Instrumentation," and LCO 3.3.5.2, "RCIC Instrumentation," and are not included in this LCO.

In general, the individual Functions are required to be OPERABLE in MODES 1, 2, and 3 consistent with the Applicability for LCO 3.6.1.1, "Primary Containment - Operating," or LCO 3.6.5.1 "Drywell," as applicable. Functions that have different Applicabilities are discussed below in the individual Functions discussion.

P8

3.3.6.1 #25

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group ~~1~~ 6 valves. P4

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 1). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits. 3.3.6.1 #41 -1

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one steam line would be able to detect the high flow. Four channels of Main Steam Line Flow—High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow—High (continued)

This Function isolates the Group ~~1~~ 6 valves. (P4)

1.d. Condenser Vacuum—Low

The Condenser Vacuum—Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum—Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Vacuum—Low Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be OPERABLE in MODES 2 and 3, when all turbine stop valves (TSVs) are closed, since the potential for condenser overpressurization is minimized. Switches are provided to manually bypass the channels when all TSVs are closed.

This Function isolates the Group ~~1~~ 6 valves. (P4)

3.3.6.1 #12

1.e. 1.f, 1.g, 1.h, 1.i, 1.j Main Steam Tunnel Ambient ~~and Differential~~ Temperature—High (P9)

Ambient ~~and Differential~~ Temperature—High is provided to detect a leak in the RCPB, and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. However, credit for these instruments is not taken in any transient or accident analysis in the ~~F~~SAR, since
U (P1)

(continued)

LAR 93-14R1

BASES

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

, 1.g, 1.h, 1.i, 1.j.

1.e, 1.f ~~Main Steam Tunnel~~ Ambient ~~and Differential~~ Temperature—High (continued) (P9)

bounding analyses are performed for large breaks such as MSLBs.

each ambient temperature

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Four channels of ~~Main Steam Tunnel Temperature—High~~ Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element. (P9)

3.3.6.1 #12

~~Eight thermocouples provide input to the Main Steam Tunnel Differential Temperature—High Function. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of four available channels.~~

The ambient ~~and differential~~ temperature monitoring Allowable Value is chosen to detect a leak equivalent to 25 gpm.

~~These~~ This Function~~s~~ isolate~~s~~ the Group ~~7~~ 6 valves. (P4)

J. ~~1.g.~~ Manual Initiation (P9)

The Manual Initiation push button channels introduce signals into the MSL isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific ~~F~~USAR safety analysis that takes credit for this Function. It is retained for ~~overall redundancy and diversity of~~ the isolation function as required by the NRC in the plant licensing basis. (P1)

3.3.6.1 #42

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

(continued)

LAR 95-14R1

BASES

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

K & J.

~~1.a.~~ Manual Initiation (continued) (P9)

Four channels of Manual Initiation Function are available and are required to be OPERABLE. ~~in MODES 1, 2, and 3, since these are the MODES in which the MSL Isolation automatic Functions are required to be OPERABLE.~~ (C1)

and Drywell — 3.3.6.1 #25

2. Primary Containment Isolation

(P17) 2.a. ~~2.a.~~ Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with isolation is implicitly assumed in the ~~F~~SAR analysis as these leakage paths are assumed to be isolated post LOCA. (P1)

3.3.6.1 #25
INSERT B150B

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

3.3.6.1 #25

(P4) This Function isolates the Group 1, 7, 8, 9, 15, and 16 ~~6A, 6B, and 7~~ valves. INSERT B150A

(P17) 2.b. ~~2.d, 2.e~~. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the automatic isolation valves ~~PCIVs~~ on high drywell pressure supports actions to ensure that offsite dose limits of

(continued)

LAR 93-14R1

INSERT B150A

The isolation of valve Group 9 also includes the actuation of the Standby Gas Treatment System, the Control Room Fresh Air System, and the containment hydrogen analyzers.

INSERT B150B

In addition, Function 2.a provides an isolation signal to certain drywell isolation valves. The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the the primary containment.

3.3.6.1 #25

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. ~~2.d, 2.f~~. Drywell Pressure—High (continued) P17

10 CFR 100 are not exceeded. The Drywell Pressure—High Function associated with isolation of the primary containment is implicitly assumed in the U FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA. P1

INSERT B151B

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

P4 This ~~These~~ Function ~~s~~ isolate ~~s~~ the Group 1, 3, and 8 ~~6A and 7~~ valves. ~~(Function 2.b), E61 isolation valves (Function 2.d), and Group 6B valves (Function 2.f).~~ INSERT B151A

~~2.c. Reactor Vessel Water Level—Low Low Low, Level 1~~

~~Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the primary containment occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.~~ P17

~~Reactor vessel water level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual wat r level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~

(continued)

LAR 93-14R1

INSERT B151A

The isolation of valve Group 8 also includes the actuation of the Standby Gas Treatment System, the Control Room Fresh Air System, and the containment hydrogen analyzers.

INSERT B151B

3.3.6T #25

In addition, Function 2.b provides an isolation signal to certain drywell isolation valves. The isolation of the drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the the primary containment.

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (P22)

2.~~g~~c. Containment ~~and Drywell Ventilation Exhaust~~ Purge Isolation Radiation—High (continued)

~~must be provided to ensure offsite dose limits are not exceeded.~~

(P4) These Functions isolate the Group ~~7~~ 8 valves.

(P17) 2.~~h~~d. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. (P1) It is retained for ~~overall redundancy and diversity of~~ the isolation function as required by the NRC in the plant licensing basis.

3.3.6.T #25

3.3.6.T #42

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are available and are required to be OPERABLE. ~~in MODES 1, 2, and 3, since these are the MODES in which the Primary Containment Isolation automatic Functions are required to be OPERABLE.~~ (C1)

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow—High

RCIC Steam Line Flow—High Function is provided to detect a break of the RCIC steam lines and initiates closure of the steam line isolation valves. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and core uncovery can occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for this Function

(continued)

LAR 93-14?1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. RCIC Steam Supply Line Pressure—Low (continued)

(TS) because of the potential for risk due to possible failure of the instruments preventing RCIC initiations.

The RCIC Steam Supply Line Pressure—Low signals are initiated from two transmitters that are connected to the system steam line. Two channels of RCIC Steam Supply Line Pressure—Low Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

3.3.6.1 #43

The Allowable Value is selected to be high enough to prevent damage to the ~~system's~~ system turbine~~s~~.

P1

This Function isolates the Group ~~6~~ 2 valves.

3.d. RCIC Turbine Exhaust Diaphragm Pressure—High

High turbine exhaust diaphragm pressure indicates that the pressure may be too high to continue operation of the associated ~~system's~~ system turbine. That is, one of two exhaust diaphragms has ruptured and pressure is reaching turbine casing pressure limits. This isolation is for equipment protection and ~~is~~ not assumed in any transient or accident analysis in the ~~FSAR~~ USAR. These instruments are included in the TS because of the potential for risk due to possible failure of the instruments preventing RCIC initiations (Ref. 3).

P1

The RCIC Turbine Exhaust Diaphragm Pressure—High signals are initiated from four transmitters that are connected to the area between the rupture diaphragms on each system's turbine exhaust line. Four channels of RCIC Turbine Exhaust Diaphragm Pressure—High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

3.3.6.1 #43

The Allowable Values are ~~high~~ Low enough to prevent damage to the ~~system's~~ system turbine~~s~~.

P4

This Function isolates the Group ~~6~~ 2 valves.

(continued)

LAR 95-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

h e

~~3.e, 3.f,~~ ~~3.i, 3.k,~~ Ambient ~~and Differential~~ Temperature—High

e

Ambient ~~and Differential~~ Temperatures are provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. These Functions are not assumed in any ~~F~~ 4 SAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks. P1

e

Ambient ~~and Differential~~ Temperature—High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each area. Six channels for RHR and RCIC Ambient Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. There are two for the RCIC room and four for the RHR area.

3.3.6.1 #12

P8 ~~There are 12 thermocouples (four for the RCIC room and eight for the RHR area) that provide input to the Area Ventilation Differential Temperature—High Functions. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of six (two for the RCIC room and four for the RHR area) available channels.~~ e

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

P4 This Function isolates the Group ~~A~~ 2 valves.

f e

~~3.g, 3.h,~~ Main Steam Line Tunnel Ambient ~~and Differential~~ Temperature—High

Ambient ~~and Differential~~ Temperature—High is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite limits may be reached. However, credit for these instruments is not taken in any

(continued)

LAR 95-14X1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

f g
~~3.g. 3.h.~~ Main Steam Line Tunnel Ambient ~~and Differential~~ Temperature—High (continued)

transient or accident analysis in the U~~F~~SAR, since bounding analyses are performed for large breaks such as MSLBs. (P1)

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Two channels of Main Steam Tunnel Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element.

3.3.6.1 #12

~~Four thermocouples provide input to the Main Steam Tunnel Differential Temperature—High Function. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of two available channels.~~ e

The Allowable Values are chosen to detect a leak equivalent to 25 gpm.

(F4) This Function isolates the Group ~~1~~ 2 valves.

3.~~i~~ g. Main Steam Line Tunnel Temperature Timer

The Main Steam Line Tunnel Temperature Timer is provided to allow all the other systems that may be leaking in the main steam tunnel (as indicated by the high temperature) to be isolated before RCIC is automatically isolated. This ensures maximum RCIC System operation by preventing isolations due to leaks in other systems. This Function is not assumed in any U~~F~~SAR transient or accident analysis; however, maximizing RCIC availability is an important function. (P1)

Two channels for RCIC Main Steam Line Tunnel Timer Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are based on maximizing the availability of the RCIC System; that is, providing

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.~~v~~ 9. Main Steam Line Tunnel Temperature Timer (continued)

sufficient time to isolate all other potential leakage sources in the main steam tunnel before RCIC is isolated.

(P4) This Function isolates the Group ~~4~~ 2 valves.

(C5) 3.~~w~~ 2. RCIC/RHR ~~High~~ Steam Line Flow—High

3.3.6.1 #12

RCIC/RHR high steam line flow is provided to detect a break of the common steam line of RCIC and RHR (steam condensing mode) and initiates closure of the isolation valves for both systems. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. Therefore, the isolation is initiated at high flow to prevent or minimize core damage. Specific credit for this Function is not assumed in any U~~F~~SAR accident or transient analysis since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC/RHR steam line break from becoming bounding. (P1)

The RCIC/RHR steam line flow signals are initiated from two transmitters that are connected to the steam line. Two channels with one channel in each trip system are available and required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Allowable Value is selected to ensure that the trip occurs to prevent fuel damage and maintains the MSLB as the boundary event.

(P4) This Function actuates the Group ~~4~~ 2 valves.

3.~~x~~ 1. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB. The RCIC isolation of the turbine exhaust is provided to prevent communication with the drywell when high drywell pressure exists. A potential leakage path exists via the turbine exhaust. The isolation is delayed until the system becomes unavailable for injection (i.e., low steam line pressure). The isolation of the RCIC turbine exhaust by Drywell Pressure—High is indirectly assumed in the U~~F~~SAR accident analysis because the turbine exhaust leakage path is not assumed to contribute to offsite doses. (P1)

(continued)

LAR 95-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.~~m~~j. Drywell Pressure—High (continued)

3.3.6.1 #12

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Two channels of RCIC Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this is indicative of a LOCA inside primary containment.

P4

This Function isolates the Group ~~7~~ 3 valves.

3.~~n~~k. Manual Initiation

INSERT 159A

3.3.6.1 #42

P5

The Manual Initiation push button channel~~s~~ introduce~~s~~s signal~~s~~ into the RCIC System isolation logic that ~~are~~ is redundant to the automatic protective instrumentation and provide~~s~~s manual isolation capability. There is no specific ~~F~~SAR safety analysis that takes credit for this Function. It is retained for ~~overall redundancy and diversity of~~ the isolation function as required by the NRC in the plant licensing basis.

P1

There ~~are four~~ is only one push button~~s~~ for RCIC~~, two~~ manual initiation ~~push buttons per~~ in a single trip system. There is no Allowable Value for this Function since the channel~~s~~ ~~are~~ is mechanically actuated based solely on the position of the push button~~s~~.

~~Four~~ Two channels of RCIC Manual Initiation are available and are required to be OPERABLE.

4. Reactor Water Cleanup System Isolation

4.a. Differential Flow—High

The high differential flow signal is provided to detect a break in the RWCU System. This will detect leaks in the RWCU System when area or differential temperature would not provide detection (i.e., a cold leg break). Should the reactor coolant continue to flow out of the break, offsite dose limits may be exceeded. Therefore, isolation of the RWCU System is initiated when high differential flow is

(continued)

LAR 93-14R1

BASES

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~4.b~~ 4.i, 4.k, 4.l

4.c, 4.d, 4.e, 4.f, 4.g, 4.h, e Ambient ~~and Differential~~ Temperature—High

Ambient ~~and Differential~~ e Temperature—High is provided to detect a leak from the RWCU System. The isolation occurs even when very small leaks have occurred and is diverse to the high differential flow instrumentation for the hot portions of the RWCU System. If the small leak continues without isolation, offsite dose limits may be reached. Credit for these instruments is not taken in any transient or accident analysis in the U~~F~~SAR, since bounding analyses are performed for large breaks such as MSLBs.

P1

Ambient ~~and differential~~ e temperature signals are initiated from temperature elements that are located in the room that is being monitored. There are eight thermocouples that provide input to the Area Temperature—High Functions (two per area). Eight channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

3.3.6.1 #12

P8

~~There are 16 thermocouples that provide input to the Differential Temperature—High Functions. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of eight available channels (two per area). Eight channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~ e

The Ambient ~~and Differential~~ e Temperature—High Allowable Values are set low enough to detect a leak equivalent to 25 gpm.

P4

These Functions isolate the Group ~~5~~ 7, 15, and 16 valves.

P14

~~4.j~~ m, ~~4.k~~ n, h Main Steam Line Tunnel Ambient ~~and Differential~~ e Temperature—High

Ambient ~~and Differential~~ e Temperature—High is provided to detect a leak in the RCPB and provides diversity to the high flow instrumentation. The isolation occurs when a very small leak has occurred. If the small leak is allowed to continue without isolation, offsite dose limits may be

(continued)

3.3.6.1 #12

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (P14)

4.j. 4.i. Main Steam Line Tunnel Ambient ~~and Differential~~ Temperature—High (continued)

reached. However, credit for these instruments is not taken in any transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

Ambient temperature signals are initiated from thermocouples located in the area being monitored. Two channels of Main Steam Tunnel Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each Function has one temperature element.

~~There are four thermocouples that provide input to the Differential Temperature—High Function. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of two available channels.~~

The Allowable Values are chosen to detect a leak equivalent to 25 gpm.

(P4) This Function isolates the Group ~~8~~ 7, 15, and 16 valves.

(P14) 4.k. Reactor Vessel Water Level—Low Low, Level 2

3.3.6.1 #12

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 2 supports actions to ensure that fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with RWCU isolation is not directly assumed in any transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs.

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water

(continued)

LAR 4-14R1

BASES

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (P14)

4.~~k~~ L. Reactor Vessel Water Level—Low Low, Level 2 (continued)

level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since the capability to cool the fuel may be threatened.

(P4) This Function isolates the Group ~~8~~ 7, 15, and 16 valves.

(P14) 4.~~l~~ J. SLC System Initiation

3.3.6.1 #12

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). SLC System initiation signals are initiated from the two SLC pump start signals.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

Two channels (one from each pump) of SLC System Initiation Function are available and are required to be OPERABLE only in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

(P4) This Function isolates the Group 7 and 10 valves.

(P14) ~~4.m.~~ K. Manual Initiation

The Manual Initiation push button channels introduce signals into the RWCU System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific ~~F~~ USAR safety analysis that takes credit for this Function. It is retained for ~~overall redundancy and diversity of~~ the isolation function as required by the NRC in plant licensing basis. (P1)

3.3.6.1 #12

(continued)

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

P14

4.m. Manual Initiation (continued)

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are available and are required to be OPERABLE. ~~in MODES 1, 2, and 3 since these are the MODES in which the RWCU System Isolation automatic Functions are required to be OPERABLE.~~ C1

P8 5. ~~Shutdown Cooling~~ RHR System Isolation

5.a. ~~5.b.~~ e Ambient ~~and Differential~~ Temperature—High

Ambient ~~and Differential~~ Temperature—High is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. ~~These Functions are~~ This function is not assumed in any UFSAR transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs. P1

3.3.6.1 #12

Ambient ~~and Differential~~ e Temperature—High signals are initiated from thermocouples that are appropriately located to protect the system that is being monitored. Two instruments monitor each area. Four channels for RHR Ambient ~~and Differential~~ Temperature—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

~~Eight thermocouples provide input to the Area Ventilation Differential Temperature—High Function. The output of these thermocouples is used to determine the differential temperature. Each channel consists of a differential temperature instrument that receives inputs from thermocouples that are located in the inlet and outlet of the area cooling system for a total of four available channels.~~ P9

(continued)

LAR 93-14R1

BASES

3.3.6.1 #12

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.a. ~~5.b.~~ e Ambient ~~and Differential~~ e Temperature—High (continued)

The Allowable Values are set low enough to detect a leak equivalent to 25 gpm. INSERT B165A C3

P4 This Function isolates the Group ~~3~~ 5 and 14 valves.

5.~~c~~ b. Reactor Vessel Water Level—Low, Level 3

3.3.6.1 #12

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level--Low, Level 3 Function associated with RHR Shutdown Cooling System isolation is not directly assumed in any transient or accident analysis, since bounding analyses are performed for large breaks such as MSLBs. The RHR Shutdown Cooling System isolation on Level 3 supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event through the 1E12*F008 and 1E12*F009 valves caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System.

3.3.6.1 #37

Reactor Vessel Water Level—Low, Level 3 signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels (two channels per trip system) of the Reactor Vessel Water Level—Low, Level 3 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (~~c~~ d) P22 3.3.6.1 #21,22 to Table 3.3.6.1-1), only two channels of the Reactor Vessel Water Level—Low, Level 3 Function are required to be OPERABLE in MODES 4 and 5 (both channels must input into the same trip system) to provide an isolation signal to the RHR Shutdown Cooling System suction from the reactor vessel, provided the RHR Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

The Reactor Vessel Water Level—Low, Level 3 Allowable Value was chosen to be the same as the RPS Reactor Vessel Water

(continued)

INSERT B165A

3.3.6.1 #12

The RHR Equipment Room Ambient ~~and Differential~~ Temperature - High Function~~s~~ ~~are~~ is only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, insufficient pressure and temperature are available to develop a significant steam leak in this piping and significant water leakage is protected by the Reactor Vessel Water Level - Low, Level 3 Function.

INSERT B165A

3.3.6.1 #12 -- The RHR Equipment Room Ambient ~~and Differential~~ Temperature - High Function~~s are~~ is only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, insufficient pressure and temperature are available to develop a significant steam leak in this piping and significant water leakage is protected by the Reactor Vessel Water Level - Low, Level 3 Function.

LAR 93-14R1

BASES

3.3.6.1 #21,22

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.c. Reactor Vessel Water Level—Low, Level 3 (continued)

b

Level—Low, Level 3 Allowable Value (LCO 3.3.1.1) since the capability to cool the fuel may be threatened. with reactor pressure less than the RHR cut-in permissive pressure

C3

The Reactor Vessel Water Level—Low, Level 3 Function is only required to be OPERABLE in MODES 3, 4, and 5 to prevent this potential flow path from lowering reactor vessel level to the top of the fuel. ~~In MODES 1 and 2, other isolations (e.g., Reactor Steam Dome Pressure—High) and administrative controls ensure that this flow path remains isolated to prevent unexpected loss of inventory via this flow path.~~

This instrumentation is required to be OPERABLE in MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure to support actions to ensure that offsite dose limits of 10CFR100 are not exceeded.

C3

This Function isolates the Group ~~3~~ 5 and 14 valves.

P4

P19 INSERT B166A

5.d. Reactor Steam Dome Pressure—High

The Shutdown Cooling System Reactor Steam Dome Pressure—High Function is provided to isolate the shutdown cooling portion of the RHR System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario and credit for the interlock is not assumed in the accident or transient analysis in the U FSAR.

P1

The Reactor Steam Dome—High pressure signals are initiated from four transmitters. Four channels of Reactor Steam Dome Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Allowable Value was chosen to be low enough to protect the system equipment from overpressurization.

This Function isolates the Group ~~3~~ 5 valves. P4

3.3.6.1 #21,22

5.e. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure—High Function associated with isolation of the RHR Shutdown Cooling System is not modeled in any UFSAR accident or transient analysis because other leakage paths (e.g., MSIVs) are more limiting.

P1

(continued)

LAR 93-14R1

INSERT B166A

3.3.6.1 #21,22

5.c. Reactor Vessel Water Level - Low Low Low, Level 1

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the shutdown cooling portion of the RHR System occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is implicitly assumed in the USAR analysis since these leakage paths are assumed to be isolated for a DBA.

Reactor vessel water level signals are initiated from level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the shutdown cooling portion of the RHR System isolates on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 10 valves.

3.3.6.1 #21,22

BASES

<STET>

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.e. Drywell Pressure—High (continued)

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure—High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

P4

This Function isolates the Group ~~3~~ 10 and 14 valves.

INSERT B167A P3

ACTIONS

B1

~~[Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.]~~

3.3.6.1 #25

and drywell

A Note has been provided to modify the ACTIONS related to primary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent ~~trains~~ divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable primary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable primary containment isolation instrumentation channel.

C4

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or 24 hours, depending on the Function, has been shown to be

(continued)

LAR 92-14-1

INSERT B167A

3.3.6.1 #2132

5.~~g~~ f. Manual Initiation

The Manual Initiation push button channels introduce signals into the RHR Shutdown Cooling System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for ~~overall redundancy and diversity of~~ the isolation function as required by the NRC in the plant licensing basis.

3.3.6.1 #42

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are available and are required to be OPERABLE.

ACTIONS

A.1 (continued)

INSERT B168A P7

acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

3.3.6.1 #25

automatic isolation valves

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSL isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that both trip systems will generate a trip signal from the given Function on a valid signal. The other isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two ~~PCIVs~~ in the associated penetration flow path can receive an isolation signal from the given Function. For Functions 1.a, 1.b, 1.d, 1.e, ~~and~~ 1.f, 1.g, 1.h, 1.i, and ~~1.j~~ e this would require both trip systems to have one channel OPERABLE or in trip. For Function 1.c, this would require both trip systems to have one channel, associated with each MSL, OPERABLE or in trip. For Functions 2.a, 2.b, ~~2.c, 2.d, 2.e~~, 3.d, 4.k, 5.c, 5.d, ~~and~~ 5.e, and 5.f, this would require one trip system to have two channels, each OPERABLE or in trip. For Functions 2.c, 3.a, 3.b, 3.c, 3.e, 3.f, 3.g, ~~3.h~~, 3.i, 3.j, ~~3.n~~, 4.a, 4.b, 4.c, ~~4.d~~, ~~4.g, 4.h, 4.i, 4.j, and 4.l~~, 4.e, 4.f, 4.g, 4.h, and 4.j this would require one trip system to have one channel OPERABLE or in trip. For ~~4.k, 4.l, 4.m, 4.n, and 4.p~~,

P9 P14 P17 P14 P14

3.3.6.1 #12

(continued)

ACTIONS

B.1 (continued)

Functions 3.j, 3.k, 4.e, 4.f, 5.a, and 5.b, each Function consists of channels that monitor several different locations. Therefore, this would require one channel per location to be OPERABLE or in trip (the channels are not required to be in the same trip system). The Condition does not include the Manual Initiation Functions (Functions 1.g, 2.f, 3.m, and 4.m), since they are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

P22 P7 P14 P3 4.k and 5.f

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and if allowed (i.e., plant safety analysis allows operation with an MSL isolated), plant operation with the MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion

(continued)

LAR 93-1001

BASES

ACTIONS

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

Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operation may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

3.3.6.1 #2

For some of the Ambient ~~and Differential~~ Temperature Functions, the affected penetration flow path(s) may be considered isolated by isolating only that portion of the system in the associated room monitored by the inoperable channel. That is, if the RWCU pump room A ambient channel is inoperable, the A pump room area can be isolated while allowing continued RWCU operation utilizing the B RWCU pump.

Alternatively, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition H must be entered and its Required Actions taken.

(continued)

LAR 93-14R1

BASES

ACTIONS

I.1 and I.2 (continued)

are provided by declaring the associated SLC subsystem inoperable or isolating the RWCU System.

3.3.6.1 #37

The Completion Time of 1 hour is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

RHR Shutdown Cooling System suction from the reactor vessel

3.3.6.1 #37

(i.e., closing either 1E12-F008 or 1E12-F009)

J.1 ~~and~~ J.2, J.3.1 and J.3.2

C13 isolated

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the ~~associated penetration flow path~~ should be ~~closed~~. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System ~~(i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated)~~. ~~ACTIONS must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.~~

provide an alternate decay heat removal capability and subsequently

C13

INSERT B172B

3.3.6.1 #45

K.1, K.2.1, ~~K.2.2~~, and ~~K.2.3~~

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be isolated (Required Action ~~K.1~~). Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable instrumentation. Alternately, the plant must be placed in a condition in which the LCO does not apply. ~~If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission production release. ACTIONS must continue until OPDRVs are suspended.~~

C5 K.1

P22

INSERT B172A

(continued)

INSERT B172A

This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

INSERT B172B

or to provide means for control of potential radioactive releases. This is accomplished by ensuring primary containment is OPERABLE. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillances may need to be performed to restore the component to OPERABLE status. In addition, at least one door in each primary containment air lock must be closed. The closed air lock door completes the boundary for control of potential radioactive releases. With the appropriate administrative controls however, the closed air lock door can be opened intermittently for entry and exit. This allowance is acceptable due to the need for containment access and due to the slow progression of events which may result from a reactor vessel draindown event. Reactor vessel draindown events would not be expected to result in the immediate release of appreciable fission products to the containment atmosphere. Actions must continue until all requirements of this Condition are satisfied.

3.3.6.1 #45

BASES (continued)

SURVEILLANCE REQUIREMENTS

B1

[Reviewer's Note: Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.] 2

As noted at the beginning of the SRs, the SRs for each Primary Containment and drywell Isolation Instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

3.3.6.1 #25

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption ~~that 6 hours is~~ of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the ~~PCIVs~~ isolation valves will isolate the penetration flow path(s) when necessary.

B12

3.3.6.1 #25

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

C8

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the ~~match~~ criteria, it may be an indication that the instrument has drifted outside its limit.

C8

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.3 (continued)

The Frequency of 92 days is based on the reliability analysis of References 5 and 6.

SR 3.3.6.1.4 and SR 3.3.6.1.5

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations ~~. Measurement and setpoint error historical determinations must be performed~~ consistent with the plant specific setpoint methodology. ~~The channel shall be left calibrated consistent with the assumptions of the setpoint methodology.~~

C9

~~If the as found setpoint is not within its required Allowable Value, the plant specific setpoint methodology may be revised, as appropriate, if the history and all other pertinent information indicate a need for the revision. The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

The Frequency of SR 3.3.6.1.4 ~~is based on the assumption of a 92 day calibration interval in the determination of equipment drift in the setpoint analysis. The Frequency of~~ and SR 3.3.6.1.5 is based on the assumption of ~~an 18 month calibration interval in the determination of~~ the magnitude of equipment drift in the setpoint analysis.

C10

3.3.6.1 #25

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3, and on drywell isolation valves in LCO 3.6.5.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

LAR 93-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.6 (continued)

Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

INSERT B176A

SR 3.3.6.1.7

MUV

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The instrument response times must be added to the PCIV closure times to obtain the ISOLATION SYSTEM RESPONSE TIME. ISOLATION SYSTEM RESPONSE TIME acceptance criteria are included in Reference 7.

3.3.6.1 #15

~~A Note to the Surveillance states that the radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This Note is necessary because of the difficulty of generating an appropriate detector input signal and because the principles of detector operation virtually ensure an instantaneous response time. Response time for radiation detection channels shall be measured from detector output or the input of the first electronic component in the channel.~~

This C11

ISOLATION SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. ~~The 18 month~~ test Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience that shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent.

REFERENCES

P1 1. UFSAR, Section [6.3]. B1

P1 2. UFSAR, Chapter [15]. B1

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

P1 4. UFSAR, Section [9.3.5]. B1

(continued)

LAR 93-14R1

INSERT B176A

3.3.6.1 #38

Testing is performed only on channels where the assumed response time does not correspond to the diesel generator (DG) start time. For channels assumed to respond within the DG start time, sufficient margin exists in the 10 second start time when compared to the typical channel response time (milliseconds) so as to assure adequate response without a specific measurement test.

10/4/94

LAR 95-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level—Low Low, Level 2 (continued)

level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the High Pressure Core Spray (HPCS)/Reactor Core Isolation Cooling (RCIC) Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.5.2, "Reactor Core Isolation Cooling (RCIC) System ~~Actuation~~ Instrumentation"), since this could indicate the capability to cool the fuel is being threatened.

C3

The Reactor Vessel Water Level—Low Low, Level 2 Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. ~~In addition, the Function is also required to be OPERABLE during CORE ALTERATIONS and operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.~~

P3

2. Drywell Pressure—High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation of high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure—High Function associated with isolation is not assumed in any UFSAR accident or transient analysis. It is retained for the ~~overall redundancy and~~

3.3.6.2 #07

F1

(continued)

LAR 95-14x1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure—High (continued)

~~diversity of the~~ secondary containment isolation instrumentation as required by the NRC approved licensing basis.

3.6.2 #37

Pressure C3

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was chosen to be the same as the ECCS Drywell Pressure—High Function Allowable Value (LCO 3.3.5.1) since this is indicative of a loss of coolant accident.

The Drywell Pressure—High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

P5

Building

3. 4. Fuel ~~Handling Area~~ Ventilation ~~and Pool Sweep~~ Exhaust Radiation—High ~~High~~

building

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refuel ~~ing floor~~ due to a fuel handling accident. When Exhaust Radiation—High ~~High~~ is detected, secondary containment isolation and actuation of the ~~SGT~~ System are initiated to limit the release of fission products as assumed in the FSAR safety analyses (Ref. 1).

associated ventilation

U P1

The Exhaust Radiation—High ~~High~~ signals are initiated from radiation detectors that are located on the ventilation exhaust piping coming from the fuel ~~handling area and the fuel handling area pool sweep, respectively.~~ The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. ~~Four~~

building ventilation

(continued)

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

P5

3, 4. Fuel ~~Handling Area~~ Building Ventilation ~~and Pool Sweep~~ Exhaust Radiation—High ~~High~~ (continued)

P5

~~channels of Fuel Handling Area Ventilation Exhaust Radiation—High High Function and four channels of Fuel Handling Area Pool Sweep Exhaust Radiation—High High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

P5

The Exhaust Radiation—High ~~High~~ Function~~s are~~ is required to be ~~OPERABLE in MODES 1, 2, and 3 where considerable energy exists, thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are required to be~~ OPERABLE during ~~CORE ALTERATIONS, OPDRVs, and~~ movement of irradiated fuel assemblies in the ~~primary or secondary containment~~ fuel building because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

5. Manual Initiation

The Manual Initiation push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the ~~overall redundancy and diversity of the~~ secondary containment isolation instrumentation as required by the NRC approved licensing basis.

P1

3.3.6.2 #37

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

(continued)

LAR 92-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.4 (continued)

C6 ~~be revised, as appropriate, if the history and all other pertinent information indicate a need for the revision. The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

C7 The Frequency is based upon the assumption of ~~an 18 month calibration interval in the determination of~~ the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing, performed on SCIVs and the ~~SGT System~~ associated ventilation subsystems in LCO 3.6.4.2, ~~and~~ LCO 3.6.4.3, LCO 3.6.4.4, LCO 3.6.4.6, and LCO 3.6.4.7, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function. D PA P5

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

P4 3.3.6.2 #25

~~SR 3.3.6.2.6~~

~~This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The instrument response times must be added to the SCIV closure times to obtain the ISOLATION SYSTEM RESPONSE TIME. ISOLATION SYSTEM RESPONSE TIME acceptance criteria are included in Reference 5.~~ D

~~A Note to the Surveillance states that the radiation detectors may be excluded from ISOLATION SYSTEM RESPONSE TIME testing. This Note is necessary because of the difficulty of generating an appropriate detector input signal and because the principles of detector operation virtually ensure an instantaneous response time. Response time for radiation detector channels shall be measured from~~

(continued)

LAR 93-14

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.6 (continued)

~~detector output or the input of the first electronic component in the channel.~~

3.3.6.2 #35

~~ISOLATION SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

This C8

REFERENCES

P1 1. UFSAR, Section [6.3]. B1

P1 2. UFSAR, Chapter [15]. B1

3. NEDO-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

4. NEDC-30851-P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentations Common to RPS and ECCS Instrumentation," March 1989.

3.3.6.2 #35

P8 5. ~~FSAR, Section [7.3].~~ RBS Technical Requirements Manual.

LAR 93-14-1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.5.3 (continued)

~~setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.~~

C6-1 The Frequency is based upon the assumption of ~~an 18 month calibration interval in the determination of~~ the magnitude of equipment drift in the setpoint analysis.

P4 SR 3.3.6.5.4

3.3.6.5 #9 -1 C1

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed for S/RVs in LCO 3.4.4, ~~LCO 3.5.1, "ECCS—Operating,"~~ and LCO 3.6.1.6 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES P1

1. UFSAR, Section 5.2.2. B1

2. UFSAR, Appendix 5A.

3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

LAR [illegible]-14R1

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Fresh Air (CRFA) System Instrumentation

BASES

BACKGROUND

The CRFA System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CRFA subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CRFA System automatically initiate action to isolate or pressurize the main control room (MCR) to minimize the consequences of radioactive material in the control room environment.

In the event of a ~~loss of coolant accident (LOCA) or a~~ Reactor Vessel Water Level—Low Low, Level 2, Drywell Pressure—High, or Control Room Ventilation Radiation Monitor signal, the CRFA System is automatically started in the isolation mode. The MCR air is then recirculated through the charcoal filter, and sufficient outside air is drawn in through the normal intake to keep the MCR slightly pressurized with respect to ~~the turbine building~~ adjacent areas. C3 P3

3.3.7.1 #21 System

The CRFA System instrumentation has two trip systems: one trip system initiates one CRFA subsystem, while the second trip system initiates the other CRFA subsystem (Ref. 1). Each trip system receives input from the Functions listed above. The Functions are arranged as follows for each trip system. The Reactor Vessel Water Level—Low Low, Level 2 and Drywell Pressure—High are arranged together in a one-out-of-two taken twice logic. The Control Room Ventilation Radiation Monitors are arranged in a two-out-of-two logic. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CRFA System initiation signal to the initiation logic.

System 3.3.7.1 #21

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CRFA System to maintain the habitability of the MCR is explicitly assumed for certain accidents as discussed in the UFSAR safety analyses (Refs. 2 and 3). CRFA System operation ensures that the radiation exposure of control room personnel, through the duration of any one of P1

3.3.7.1 #21 System

(continued)

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A.

CRFA instrumentation satisfies Criterion 3 of the NRC Policy Statement. System 3.3.7.1 #21

The OPERABILITY of the CRFA instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

System 3.3.7.1 #21

Allowable Values are specified for each CRFA Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. These nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint that is less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES (continued)

ACTIONS

L1

Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.

A Note has been provided to modify the ACTIONS related to CRFA System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent ~~trains~~ divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CRFA System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CRFA System instrumentation channel.

CS

3.3.7.1 #21

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

3.3.7.1 #21

B.1 and B.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CRFA System design, an allowable out of service time of 24 hours has been shown to be acceptable (Refs. 4 and 5) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CRFA System initiation capability. A Function is considered to be maintaining CRFA System initiation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate an initiation signal from the given Function on a valid signal. This would

(continued)

LAR 93-14R1

BASES

3.3.7.1 #21

ACTIONS

B.1 and B.2 (continued)

subsystems System

require one trip system to have two channels, each OPERABLE or in trip. In this situation (loss of CRFA initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate. If the Function is not maintaining CRFA initiation capability, ~~the~~ CRFA ~~System~~ must be declared inoperable within 1 hour of discovery of loss of CRFA initiation capability in both trip systems. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Actions taken.

C6 both

3.3.7.1 #21

C.1 and C.2

System

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CRFA design, an allowable out of service time of 12 hours has been shown to be acceptable (Refs. 4 and 6) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CRFA initiation capability. A Function is considered to be maintaining CRFA initiation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate an initiation signal from the given Function on a valid signal. This would require one trip system to have two channels, each OPERABLE or in trip. In this situation (loss of CRFA initiation capability), the 12 hour allowance of Required Action C.2 is not appropriate. If the Function is not maintaining CRFA initiation capability, ~~the~~ CRFA ~~System~~ must be declared inoperable within 1 hour of discovery of loss of CRFA initiation capability in both trip systems. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition, per Required Action C.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to

both C6 subsystems

(continued)

LAR 93-14R1

ACTIONS

C.1 and C.2 (continued)

accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Actions taken.

D.1 and D.2

3.3.7.1 #21 System

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CRFA design, an allowable out of service time of 6 hours is provided to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CRFA initiation capability. A Function is considered to be maintaining CRFA initiation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate an initiation signal from the given Function on a valid signal. This would require one trip system to have two channels, each OPERABLE or in trip. In this situation (loss of CRFA initiation capability), the 6 hour allowance of Required Action D.2 is not appropriate. If the Function is not maintaining CRFA initiation capability, ~~the~~ CRFA ~~System~~ must be declared inoperable within 1 hour of discovery of loss of CRFA initiation capability in both trip systems. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition, per Required Action D.2. Placing the inoperable channel in trip performs the intended function of the channel (starts the associated CRFA subsystem in the isolation mode). Alternately, if it is not desired to place the channel in trip (e.g., as in the case where it is not desired to start the subsystem), Condition E must be entered and its Required Actions taken.

3.3.7.1 #26 both subsystems

The 6 hour Completion Time is based on the consideration that this Function provides the primary signal to start the CRFA System, thus ensuring that the design basis of the CRFA System is met.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

With any Required Action and associated Completion Time not met, the associated CRFA subsystem must be placed in the isolation mode of operation (Required Action D.1) to ensure that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the CRFA subsystem in operation must provide for automatically reinitiating the subsystem upon restoration of power following a loss of power to the CRFA subsystem(s). ~~As noted, if the toxic gas protection instrumentation is concurrently inoperable, then the CRFA subsystem shall be placed in the toxic gas mode instead of the isolation mode. This provides proper protection of the control room personnel if both toxic gas instrumentation (not required by Technical Specifications) and radiation instrumentation are concurrently inoperable.~~ Alternately, if it is not desired to start the subsystem, the CRFA subsystem associated with inoperable, untripped channels must be declared inoperable within 1 hour.

6

The 1 hour Completion Time is intended to allow the operator time to place the CRFA subsystem in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, or for placing the associated CRFA subsystem in operation.

SURVEILLANCE REQUIREMENTS

~~[Reviewer's Note: Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.]~~

As noted at the beginning of the SRs, the SRs for each CRFA Instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

System

3.3.7.1 #21

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CRFA initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the

(continued)

LAR 93-14R1

BASES

C10 of

SURVEILLANCE REQUIREMENTS (continued)

applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 4, 5, and 6) assumption ~~that 6 hours is~~ the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CRFA will initiate when necessary.

3.3.7.1 #21 System

SR 3.3.7.1.1

normally C7

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the indicated parameter for one instrument channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

C7

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the ~~match~~ criteria, it may be an indication that the instrument has drifted outside its limit.

C7

The Frequency is based upon operating experience that demonstrates channel failure is rare. ~~Thus, performance of the CHANNEL CHECK ensures that undetected outright channel failure is limited to 12 hours.~~ The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

C8

(continued)

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
TS: 3.3.3.1 - POST ACCIDENT MONITORING INSTRUMENTATION

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording preferences revised to reflect appropriate plant specific requirements.

5.0 #131 The detail regarding special reports are moved to this section consistent with revisions proposed in Section 5.0.

PLANT SPECIFIC DIFFERENCE

P.1 This comment number is not used for this station.

P.2 ~~The correct plant specific reference is provided for consistency with revisions proposed in Section 5.8.~~

P.3 The plant specific Type A and Category 1 PAM instruments are identified in accordance with the NUREG Reviewer's Note.

P.4 The current required allowed out-of-service time and CHANNEL CALIBRATION frequency for the Hydrogen Analyzers is retained.

P.5 The current plant specific number of channels of suppression pool monitoring required is retained.

CHANGE/IMPROVEMENT TO NUREG STS

C.1 This comment number is not used for this station.

C.2 The Bases are corrected. The reference Specification in the Administrative Controls Section does not require the Special Reports to be approved by any specific individual or committee.

C.3 Condition G is not applied to reactor vessel water level in the LCO. Therefore, the Bases are revised to match the LCO.

C.4 These changes are proposed to revise specific terminology to that which is generically preferred for application to the BWR/6 plants. The BWR LCOs do not use the term "train", however, "division" is used in several places.

C.5 This comment number is not used for this station.

LAR 93-14R1

RIVER BEND

SECTION 3.4

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R7

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.4 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LCO 3.4.1
LCO 3.4.2
LCO 3.4.11

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

~~3.4.1.1~~ The reactor coolant system recirculation loops shall be in operation and in Region I as specified in Figure 3.4.1.1-1 with either:

3.4 #17

a. Two recirculation loops operating with limits and setpoints per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, 3.3.6, or A1

b. A single loop operating with:

LA1

1. Volumetric recirculation loop flow rate less than or equal to 33,000 gpm, and

2. The recirculation loop flow control system in the loop Manual (Position Control) Mode, and

3.4 #17

3. THERMAL POWER less than or equal to 70% of RATED THERMAL POWER and

4. Limits and setpoints for single recirculation loop operation per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, and 3.3.6. A1 LA2

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*

ACTION

a. During single loop operation, with volumetric recirculation loop flow rate greater than 33,000 gpm, immediately initiate corrective action to reduce flow to less than or equal to 33,000 gpm within 1 hour. LA1

3.4 #17

b. During single loop operation, with the recirculation flow control system not in the Loop Manual mode, immediately initiate corrective action to place the recirculation flow control system in the Loop Manual mode within 1 hour. LA1

LCO 3.4.1 Act B

c. During single loop operation, with THERMAL POWER greater than 70% of RATED THERMAL POWER, immediately initiate corrective action to reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER within 1 hour.

LCO 3.4.1 NOTE

d. Within ~~4~~ 12 L1 hours upon entry into single loop operation, verify that the operating limits in Specification 3.2.1 A1 have been appropriately adjusted for single loop operation.

*See Special Exception 3.10.4 A3

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LAR - 3-14R1

3.4 #17

INSERT COND F — M1

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 NOTE — e. Within 12 hours upon entry into single loop operation, verify that the setpoints in Specifications 2.2.1, 3.2.2 and 3.3.6 are within appropriate limits. (A1)

LCO 3.4.11 COND A — f. During single loop operation with either THERMAL POWER ≤ 30% of RATED THERMAL POWER or recirculation loop flow in the operating loop is ≤ 50% of rated recirculation loop flow and temperature differences exceeding the limits in Surveillance Requirement 4.4.1.1.4, suspend THERMAL POWER or recirculation loop flow increases.* (LA3)

3.4 #17

LCO 3.4.1 COND C+D — g. With one or two reactor coolant system recirculation loops in operation and total core flow greater than 39% and less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Region II of Figure 3.4.1.1-1: (LA1)

LCO 3.4.1 COND C — 1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.2):

a) At least once per 8 hours, and

b) Within 30 minutes after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

3.4 #17

LCO 3.4.1 COND D — 2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow to greater than or equal to 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Region II of Figure 3.4.1.1-1.

LA1

LCO 3.4.1 COND E — h. With one or two reactor coolant system recirculation loops in operation and total core flow less than 39% of rated core flow and THERMAL POWER greater than the limit specified in Region III of Figure 3.4.1.1-1, immediately within 15 minutes initiate corrective action to increase core flow to greater than or equal to 39% of rated core flow or reduce THERMAL POWER to less than the limit specified in Region III of Figure 3.4.1.1-1 within 4 hours.

3.4 #17

(LA3) *With one recirculation loop not in operation and isolated, the differential temperature requirements of Surveillance Requirement 4.4.1.1.4b and c are not applicable, and the provisions of Specification 3.0.4 are not applicable with respect to Surveillance Requirement 4.4.1.1.4b and c.

(LA1) **Detector levels A and C of one LPRM string in the center of the core should be monitored.

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LAR 93-14R1

SURVEILLANCE REQUIREMENTS

INSERT NEW LCO 3.4.2 A4

~~4.4.1.1.1~~ Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

SR 3.4.2.1 ~~a~~. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic control unit, and

SR 3.4.2.2 ~~b~~. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing

LA4 COND ~~B~~ C COND ~~C~~ D

4.4.1.1.2 Establish a baseline APRM and LPRM* neutron flux noise valve within the regions for which monitoring is required (Specification 3.4.1.1 ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

3.4 #17

LA1

4.4.1.1.3 Initially, within 1 hour upon entry into single loop operation and once per 12 hours thereafter, verify that:

a. THERMAL POWER is less than or equal to 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Loop Manual (Position Control) mode, and

c. The volumetric recirculation flow rate is less than or equal to 33,000 gpm.

SR 3.4.11.8 & SR 3.4.11.9

NOTE

~~4.4.1.1.4~~ With one reactor coolant system recirculation loop not in operation, and either THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated recirculation loop flow, within 15 minutes prior to an increase in THERMAL POWER or recirculation loop flow, verify that the following differential temperature requirements are met:

SR 3.4.11.3 ~~a~~. ≤ 100°F (LA1) between reactor vessel steam space coolant and bottom head drain line coolant, and

LA1

*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of core should be monitored.

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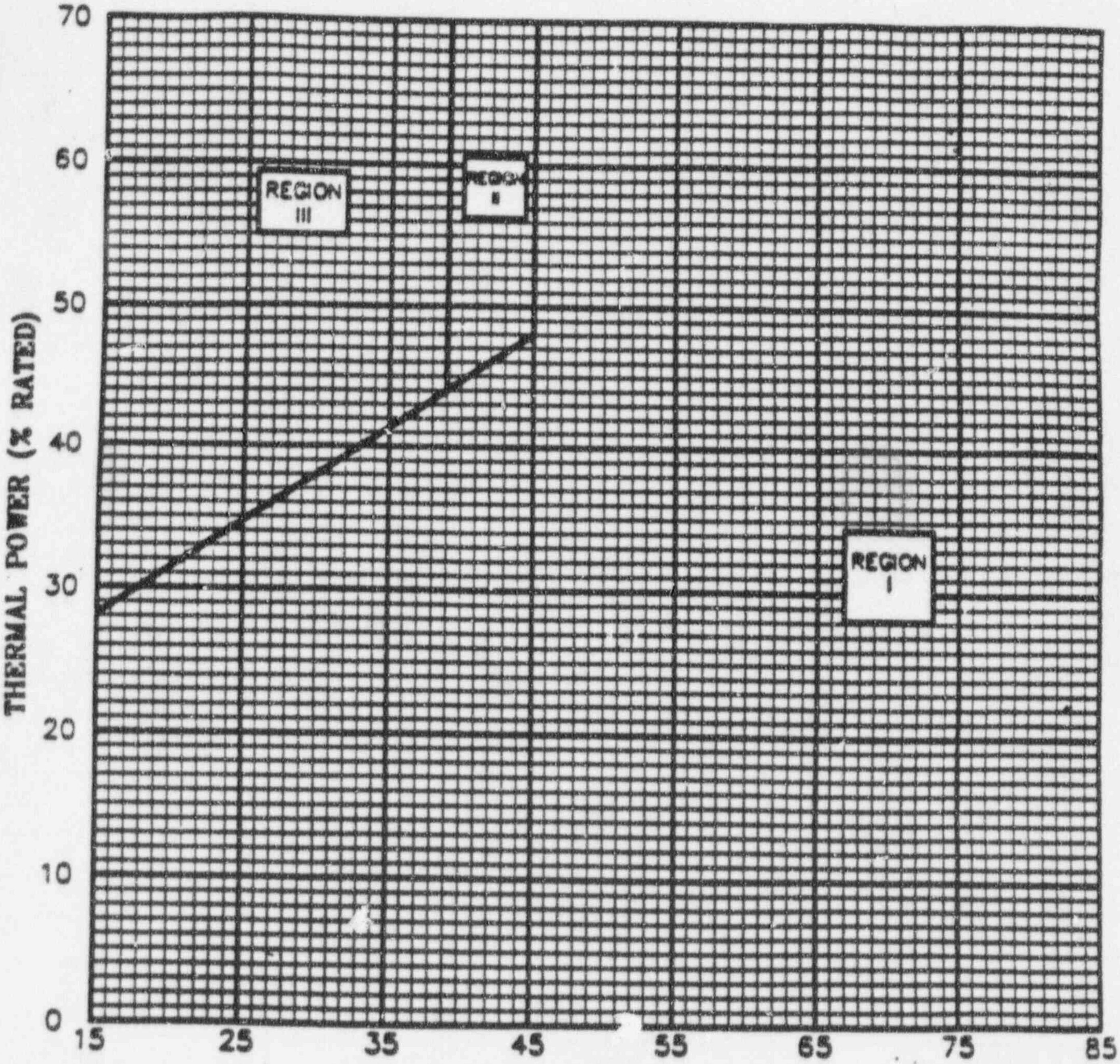
LAR 93-14R1

3.4 #17



CORE FLOW (% RATED)
FIGURE 3.4.1.1-1
THERMAL POWER VERSUS CORE FLOW

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LAR 95-14R1

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

<LCO 3.4.1>

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

Recirculation loop flow mismatch shall be maintained within:

SR 3.4.1.1

~~a.~~ 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

~~b.~~ 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With recirculation loop flows different by more than the specified limits, either:

COND A A3 a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

3.4 #17 b. Shutdown one of the recirculation loops and take the ACTION required by Specification 3.4.1.1.** A1

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

L2 INSERT SR 3.4.1.1 NOTE

A2 *See Special Test Exception 3.10.4.

**The provisions of Specification 3.0.4 are not Applicable. A1

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REACTOR COOLANT SYSTEM

<LCO 3.4.8>

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.8

~~3.4.5~~ The specific activity of the primary coolant shall be limited to:

3.4 #56

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

~~b. Less than or equal to 100/Ē microcuries per gram.~~ L3

APPLICABILITY: OPERATIONAL CONDITIONS 1, ~~2, 3 and 4~~.

L1 and, 2 and 3 with any main steam line not isolated.

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant; INSERT COND A NOTE L4

Req. Act A.2

Cond B

~~1.~~ Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN, with the main steam line isolation valves closed, within 12 hours.

~~2. Greater than 100/Ē microcuries per gram, be in at least HOT SHUTDOWN, with the main steam line isolation valves closed, within 12 hours.~~ L3

Req. Act A.1

Req. Act B.1

~~b.~~ In OPERATIONAL CONDITIONS 1, 2, 3 ~~or 4~~, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 ~~or greater than 100/Ē microcuries per gram~~, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. L1 L3 L2

A1

~~c.~~ In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour*, or

Req. Act A.1 &

Req. Act B.1

2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restore to within its limit.

*Not applicable during the startup test program. A2

LAR 93-14?1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

~~4.4.5~~ The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

SR 3.4.8.1
Req. Act A.1
Req. Act B.1

3.4
#56

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
~~1. Gross Beta and Gamma Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3~~ L3
SR 3.4.8.1 ~~2~~. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per ~~31~~ 7 days M1	1
~~3. Radiochemical for Ē Determination~~	~~At least once per 6 months*~~	~~1~~ L3
~~4~~. Isotopic Analysis for Iodine Req. Act A.1 & Req. Act B.1	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4# L1
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2 A1
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1 LA1

3.4 #56

L3

~~*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.~~

#Until the specific activity of the primary coolant system is restored to within its limits.

Req. Act A.1 & B.1

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11

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

<LCO 3.4.11>

3.4 #72

LIMITING CONDITION FOR OPERATION

LCO 3.4.11

~~3.4.6.1~~ The reactor coolant system temperature and pressure 3.4.11-1 shall be limited in accordance with the limit lines shown on Figure ~~3.4.6.1-1~~ (1) curves A and A' for hydrostatic or leak testing; (2) curves B and B' for heatup by non-
SR 3.4.11.2 nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C and C' for operations with a critical core other than low power PHYSICS TESTS, with:

SR 3.4.11.1
a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

LA1
c. A maximum temperature change of 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

SR 3.4.11.6 SR 3.4.11.7
d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

INSERT Cond A NOTE - A4

ACTION:

COND A/C
With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes (A1); perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; and determine that the reactor coolant system remains acceptable for continued operations. (M1) Otherwise, be in at least HOT
COND B SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

~~4.4.6.1.2~~ During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined, at least once per 30 minutes, to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C', as applicable. SR 3.4.11.1 NOTE

3.4 #72

3.4 #72

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.4.11.2

~~4.4.6.1.2~~ The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure ~~3.4.6.1-1~~ 3.4.11-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to SR 3.4.11.1 bring the reactor to criticality and at least once per 30 minutes during system heatup.

A3 4.4.6.1.3 The reactor vessel material surveillance specimens A6 shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H in accordance with the schedule in ~~Table 4.4.6.1.3-1~~. The results of these examinations shall be used to update the curves of A6 Figure 3.4.6.1-1.

3.4 #72

SR 3.4.11.5

~~4.4.6.1.4~~ The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

SR 3.4.11.7 ~~1~~. ≤ 100°F, at least once per 12 hours.

SR 3.4.11.6 ~~2~~. ≤ 80°F, at least once per 30 minutes.

SR 3.4.11.5 NOTE b. Within 30 minutes prior to and A2 at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

A5 INSERT SR 3.4.11.6 NOTE & SR 3.4.11.7 NOTE

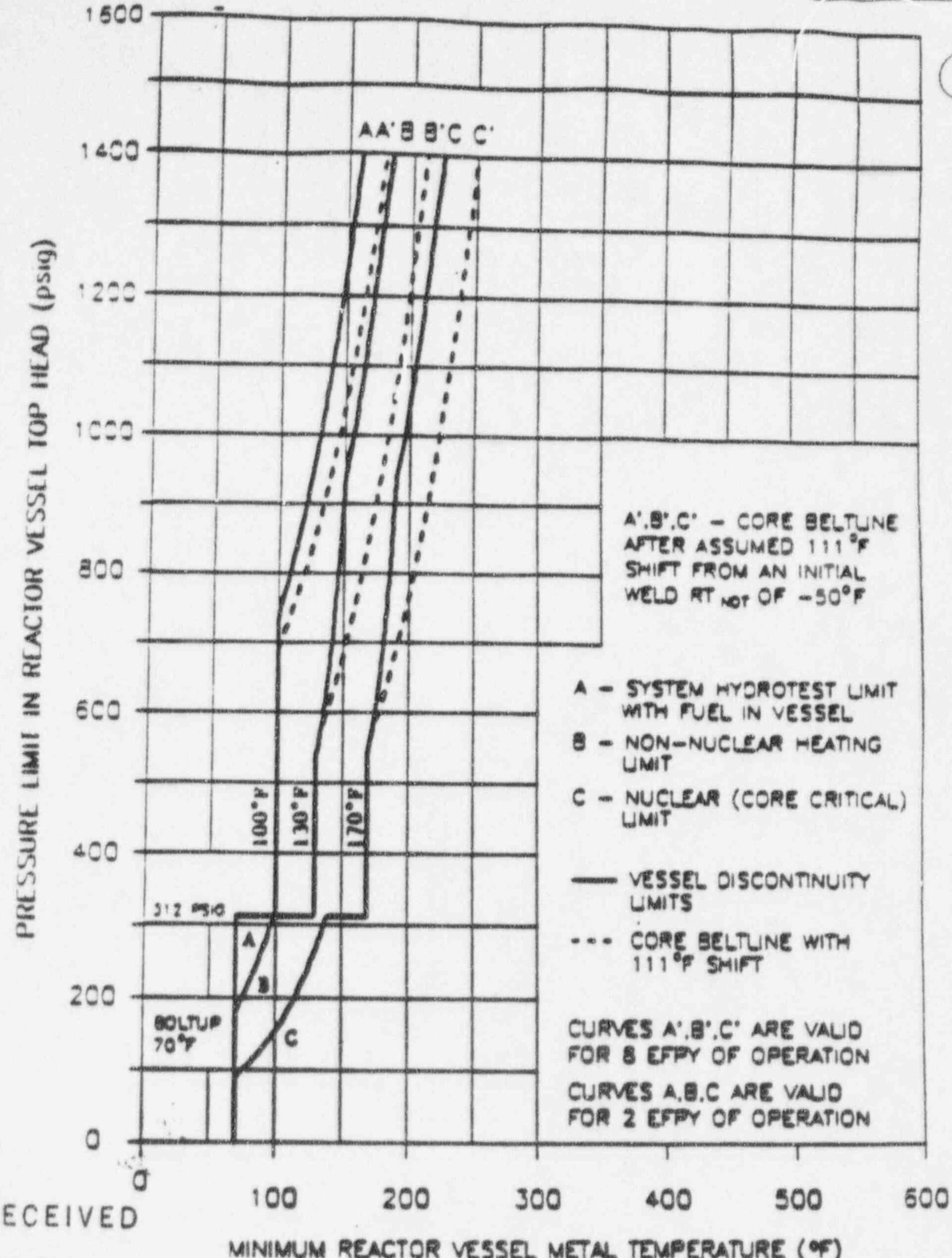

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FIGURE ~~3.4.6.1-1~~ 3.4.11-1
MINIMUM TEMPERATURE REQUIRED VS ~~REACTOR~~ RCS PRESSURE

LAR 93-14 R1

ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.4.1.1 - RECIRCULATION LOOPS

ADMINISTRATIVE

A.1 The format of the proposed Technical Specifications does not include providing "cross references." Each LCO stands alone and adequately prescribes the use of its requirements without such references. Therefore the existing reference to the other specifications serves no functional purpose, and its removal is purely an administrative difference in presentation.

3.4 #17

A.2 ~~The existing Actions to "restore...within (2 or 4) hours" is proposed to be revised to "initiate action to restore ...Immediately" for operation in restricted power/flow Region III. The existing requirement would appear to provide a time period during which power and flow requirements could exceed the limits, even if capable of being returned to within limits. Also, if the parameters are incapable of being restored to within the limits within 4 hours, the existing action would appear to result in the requirement for an LER. The intent of the action is believed to be more appropriately presented in proposed Required Action C.1. This interpretation of the intent is supported by the BWR Standard Technical Specification, NUREG-1434. As an enhanced presentation of the existing intent, the proposed change is deemed to be administrative.~~

Repeat

A.3 The format of the proposed Technical Specifications does not include providing "cross references." LCO 3.0.7 adequately prescribes the use of the Special Operations LCOs without such references. Therefore the existing reference to the Special Test Exception(s) serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.4 A new LCO, including appropriate ACTIONS, is proposed to clarify the intent for OPERABILITY of the recirculation system flow control valves. These valves are currently included in the Recirculation Loops specification by required surveillances. The proposed change would provide only additional clarification of the current requirements, and is therefore considered an administrative change.

A.5 Thermal stresses on vessel components are dependent on the temperature difference between the idle loop coolant and the RPV coolant. Proposed SR 3.4.11.9 ensures the temperature difference between the idle loop and the RPV coolant is acceptable. A requirement to monitor the temperature difference between an idle loop and an operating loop is unnecessary and can be deleted as it is redundant to the loop-to-coolant requirement of SR 3.4.11.9. However, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature."

A.6 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.4.1.3 - RECIRCULATION LOOP FLOW

ADMINISTRATIVE

A.1 The format of the proposed Technical Specifications does not include providing "cross references." LCO 3.4.1 adequately prescribes the necessary conditions for compliance without such references. Therefore the existing reference to "take the ACTION required by Specification 3.4.1.1**" serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.2 The format of the proposed Technical Specifications does not include providing "cross references." LCO 3.0.7 adequately prescribes the use of the Special Operations LCOs without such references. Therefore the existing reference to the Special Test Exception(s) serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.3 The revised presentation of actions (based on the BWR Standard Technical Specifications, NUREG-1434) does not propose to explicitly detail options to "restore...to OPERABLE status." This action is always an option, and is implied in all Conditions. Omitting this action is purely editorial.

A.4 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

TECHNICAL CHANGE - LESS RESTRICTIVE

3.4 #17

"Generic" This comment number is not used for this station.

LA.1 ~~The Required Action to "shutdown" one recirculation loop is relocated to procedures. Shut down of the loop is not always necessary, but may be preferred under some conditions. Relocating the method to meet the requirements to procedures provides additional flexibility, when conditions allow, and maintains the current action as an option.~~ e

LA.2 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.4.5 - SPECIFIC ACTIVITY

ADMINISTRATIVE

A.1 Increased sampling is already required when the LCO limit is exceeded (Required Actions A.1 and B.1). Therefore, no additional surveillance requirements are necessary as an action.

A.2 The one time exception for the startup test program is no longer needed and can be omitted.

3.4 #56

A.3 This comment deleted from LAR 93-14 based on deletion of E-bar requirements per BWR-12 C1.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

3.4 #56

M.1 This proposed change modifies Item 2 of Table 4.4.5-1 to change the frequency for isotopic analysis for dose equivalent I-131 concentration from at least once per 31 days to at least once per seven days.

Modification of Table 4.4.5-1, Item 2, to require isotopic analysis for dose equivalent I-131 concentration at least once per seven days, as opposed to the current requirement of at least once per 31 days, provides a compensatory measure for ensuring that even with deletion of the requirement that gross specific activity remain less than or equal to 100/E-bar μCi/gram, offsite doses will remain within a small fraction of the limits of 10 CFR 100.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 The offgas isotopic analysis for xenon and krypton are not direct measurements related to the LCO limits. They are used to routinely monitor and trend coolant activity and applicable to plant specific controls and administrative limits only. Therefore, this surveillance has been relocated to plant specific administrative controls.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.4.5 - SPECIFIC ACTIVITY

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

"Specific"

L.1 The Applicability is limited to those conditions which represent a potential for release of significant quantities of radioactive coolant to the environment. Mode 4 is omitted since the reactor is not pressurized and the potential for leakage is significantly reduced. In Modes 2 and 3, with the main steam lines isolated, no escape path exists for significant releases and requirements for limiting the specific activity are not required. The Required Actions are also modified to reflect the new Applicability, and an option for exiting the applicable Modes is provided for cases where isolation is not desired.

L.2 The requirement to conduct isotopic analysis for iodine when the gross specific activity limits are exceeded is deleted. Unless conditions exist that indicate that iodine is beyond limits, the increased frequency for analysis of iodine is unnecessary in that it provides no useful information with regard to the noncompliance.

L.3 This proposed change deletes TS LCO 3.4.5.b, associated Actions and Surveillance Requirements which requires gross specific activity for non-iodines in the reactor coolant to be limited to less than or equal to 100/E-bar μCi/gram. This proposed change also deletes Item 1 of Table 4.4.5-1 requiring gross beta and gamma activity determination at least once per 72 hours.

3.4 #50

The Bases for TS 3.4.5 state that the intent of the requirement to limit the specific activity of the reactor coolant is to ensure that whole body and thyroid doses at the site boundary would not exceed a small fraction of the limits stated in 10 CFR 100 (i.e., 10% of 25 rem and 300 rem, respectively) in the event of a main steam line failure outside containment. To ensure that offsite thyroid doses do not exceed 30 rem, reactor coolant dose equivalent iodine-131 (DEI) is limited to less than or equal to 0.2 μCi/gram. Likewise, reactor coolant gross specific activity is limited to less than or equal to 100/E-bar μCi/gram to ensure that whole body doses to not exceed 2.5 rem.

LCO 3.11.2.7 (ITS LCO 3.7.4) associated with radioactive effluents requires that the gross gamma radioactivity rate of the noble gases Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88 measured prior to the holdup pipe be limited to less than or equal to 290 millicuries/second. The Bases for LCO 3.11.2.7 state that restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total-body exposure to an individual at the exclusion

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DISCUSSION OF CHANGES
CTS: 3.4.5 - SPECIFIC ACTIVITY

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

area boundary will not exceed a small fraction of the limits of 10 CFR 100 in the event this effluent is inadvertently discharged directly to the environment without treatment.

3.4 #56

The offgas treatment system, as required by LCO 3.11.2.7 (ITS LCO 3.7.4), provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100 in the event of a main steam line failure. Therefore, LCO 3.4.5.b is redundant and places an unnecessary burden on the licensee without a commensurate increase in the margin of safety. Elimination of LCO 3.4.5.b will allow plant personnel to focus attention on efficient, safe operation of the plant without the unnecessary distraction of the redundant surveillance requirement. Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per seven days.

Since (1) the reactor coolant limit on DEI adequately assures that offsite doses will not exceed small fractions of the limits of 10 CFR 100 in the event of a main steam line failure outside containment and (2) gross gamma radioactivity rate of the noble gases measured prior to the holdup pipe is limited by LCO 3.11.2.7 (ITS LCO 3.4.7) to a value that provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100, the requirements associated with LCO 3.4.5.b are unnecessary.

L.4 A Note is added to the Required Actions for Condition A to indicate that LCO 3.0.4 is not applicable. Entry into the Applicable Modes should not be restricted since the most likely response to the condition is restoration of compliance within the allowed 48 hours. Further, since the LCO limits assure the dose due to a LOCA would be a small fraction of the 10 CFR 100 limit, operation during the allowed time frame would not represent a significant impact to the health and safety of the public.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.4.6.1 - RCS PRESSURE/TEMPERATURE LIMITS

ADMINISTRATIVE
(continued)

A.5 A Note is provided to clarify the current intent of allowing entry into the applicable MODES without having performed this surveillance requirement. Since this requirement is only performed during the specified conditions, this change is consistent with current application and is considered administrative.

A.6 Changes to this section are being concurrently proposed in a Technical Specification amendment for relocation of component lists. The No Significant Hazards Consideration statement made in the referenced letter is still applicable to these changes. As such this change is considered administrative in this submittal.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 A specific completion time for the engineering evaluation is proposed. The proposed time of 72 hours is considered reasonable for operation in Modes 1, 2 and 3 because the limits represent controls on long term vessel fatigue and usage factors. In Modes 4 and 5, the proposed frequency would prevent entry in the operating modes which is consistent with the current LCO 3.0.4. Additionally, exceeding the P/T limits for less than 30 minutes is not expected to present an immediate threat to the RCS integrity.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 ~~The~~ This specific limits e for reactor coolant system pressure and temperature have been relocated to a plant specific controlled document ~~, e.g., a Pressure-Temperature Limits Report (PTLR)~~. The design features and system operational limits are also described in the USAR. ~~Changes to the PTLR will be controlled by the provisions of the proposed PTLR controls in Chapter 5 of the Technical Specifications.~~ e

3.4 #72

LAR 93-14R1

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

(LAR 93-14R1)

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.4.5 - SPECIFIC ACTIVITY

"L3" CHANGE

Entergy Operations, Inc. (EOI) has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   3.4 #56

   The proposed change deletes Technical Specification (TS) Limiting Condition of Operation (LCO) 3.4.5.b which requires that the reactor coolant gross specific activity remain less than or equal to 100/E-bar μCi/gram and the requirement to determine E-bar at least once per six months, Table 4.4.5-1, Item 3. The proposed change also deletes Action a.2 associated with LCO 3.4.5.b that requires the plant to be in hot shutdown with the main steam isolation valves closed within 12 hours after the reactor coolant gross specific activity exceeds 100/E-bar μCi/gram. Additionally, the proposed change involves some administrative changes to support the deletion of LCO 3.4.5.b.

   BWR operating experience has shown that as fuel leakage increases, dose equivalent iodine-131 (DEI) approaches the TS limit much more rapidly than does the gross specific activity. The BWR design utilizes main condenser air ejectors to remove non-condensable gases from the reactor coolant. The non-condensable gases are then sampled, monitored, and processed by the offgas system prior to release to the environment. The offgas pretreatment sample provides a more representative sample of the noble gases that would be released in the event of a main steam line failure outside containment than does the reactor coolant sample currently being taken from the reactor recirculation system. The offgas pretreatment monitor includes a setpoint which responds to release rates above a specified level which is established to ensure that untreated releases would not result in a whole body dose that exceeds a small fraction of the limits of 10 CFR 100. The sample point on the reactor recirculation system currently being used to collect information regarding gross specific activity will continue to be available for use in the event of a main steam line failure upstream of the offgas treatment system.

   The Bases for TS 3.4.5 state that the intent of the requirement to limit specific activity in the reactor coolant is to ensure that the whole body and thyroid doses at the site boundary will not exceed a small fraction of the limits specified in 10 CFR 100 (i.e., 10 percent of 25 rem and 300 rem, respectively) in

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the event of a main steam line failure outside containment. To ensure that offsite thyroid doses do not exceed 30 rem, reactor coolant DEI is limited to less than or equal to 0.2 μCi/gram. Likewise, reactor coolant gross specific activity is limited to less than or equal to 100/E-bar μCi/gram to ensure that offsite whole body doses do not exceed 2.5 rem. Reactor coolant gross specific activity is not an initiator of any accident evaluated in the USAR and therefore, deletion of LCO 3.4.5.b which limits reactor coolant gross specific activity to a value less than or equal to 100/E-bar μCi/gram will not result in an increase in the probability of an accident previously evaluated in the USAR.

3.4 #56

LCO 3.11.2.7 (ITS LCO 3.7.4) associated with radioactive effluents requires that the gross gamma radioactivity rate of the noble gases Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88 measured by the main condenser offgas system pretreatment monitor be limited to less than or equal to 290 millicuries/second. The Bases for LCO 3.11.2.7 state that restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of the 10 CFR 100 in the event this effluent is inadvertently discharged directly to the environment without treatment.

The offgas treatment system, as required by LCO 3.11.2.7 (ITS LCO 3.7.4), provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100 in the event of a main steam line failure. Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per seven days. Since the proposed change will ensure that the offsite doses resulting from a main steam line failure will continue to be limited to a small fraction of the 10 CFR limits the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical modification to the plant or to plant operation. The reactor coolant gross specific activity is a parameter that is monitored to prevent offsite doses from exceeding a small fraction of the 10 CFR 100 limits and support calculation of offsite doses in the event of a main steam line failure outside containment. As such, the reactor coolant specific activity is utilized to mitigate the radiological consequences of a main steam line failure and is not considered to be an initiator for any accident.

3.4
& 5b

Additionally, the offgas treatment system will provide an equal or better means for monitoring the reactor coolant gross specific activity than would the reactor recirculation system currently being used for this purpose. In the event of a main steam line break upstream of the condenser that would prevent use of the offgas treatment system to monitor reactor coolant gross specific activity, the existing sample point on the reactor recirculation system would continue to be available. Accordingly, deletion of the requirement to limit reactor coolant gross specific activity will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The Bases for TS 3.4.5 state that the intent of the requirement to limit specific activity in the reactor coolant is to ensure that the whole body and thyroid doses at the site boundary will not exceed a small fraction of the limits specified in 10 CFR 100 (i.e., 10 percent of 25 rem and 300 rem, respectively) in the event of a main steam line failure outside containment.

As stated above, LCO 3.11.2.7 associated with radioactive effluents requires that the gross gamma radioactivity rate of the noble gases Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88 measured by the main condenser offgas system pretreatment monitor be limited to less than or equal to 290 millicuries/second. The Bases for LCO 3.11.2.7 state that restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total-body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of the 10 CFR 100 in the event this effluent is inadvertently discharged without treatment directly to the environment.

The offgas treatment system, as required by LCO 3.11.2.7 (ITS LCO 3.7.4), provides reasonable assurance the reactor coolant gross specific activity is maintained at a level sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100 in the event of a main steam line failure. Therefore, LCO 3.4.5.b is redundant and places an unnecessary burden on the licensee without a commensurate increase in the margin of safety. Elimination of LCO 3.4.5.b will allow plant personnel to focus attention on efficient, safe operation of the plant without the distraction of an unnecessary surveillance requirement. Accordingly, the proposed change enhances operation of the plant without reducing the margin of safety associated with a main steam line

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3.4 #56

failure outside of containment (i.e., offsite doses remain a small fraction of the 10 CFR 100 limits).

Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per seven days. Therefore, the proposed change does not result in a significant reduction in a margin of safety.

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.4 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

3.4 #17

modified

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

C1 3. Required limits ~~and setpoints modifications~~ for single recirculation loop operation as specified in the COLR; and

LCO 3.4.1 A. Two recirculation loops ~~with matched flows~~ shall be in operation, ~~and operating in Region I~~,

with:
1. Matched flows; and
2. Total core flow and THERMAL POWER within limits.

[OR

B. One recirculation loop ~~may~~ shall (C1) be in operation ~~provided the~~ with: and
~~following limits are applied when the associated LCO is applicable:~~

~~a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits [specified in the COLR];~~

~~b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits [specified in the COLR]; and~~

B1

1. THERMAL POWER ≤ 70% RTP;
2. Total core flow and THERMAL POWER within limits;

~~c.~~ ~~b.~~ 4 LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—High), Allowable Value of Table 3.3.1.1-1 ~~is~~ reset for single loop operation.]

INSERT 1A — P2

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A.* Requirements of the LCO not met.~~	~~A.1 Satisfy the requirements of the LCO.~~	~~24 hours~~

~~* Pending resolution of stability issue.~~

(continued)

P2 — INSERT 1B

LAR 93-14R1

INSERT 1A

-------------------------NOTE-------------------------
Required limit and setpoint modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

3.4 #17

INSERT 1B

A. Recirculation loop jet pump flow mismatch not within limits.	A.1 Shutdown one ~~Restore the~~ recirculation loop ~~flows to within limits.~~	2 hours
C ~~B.~~ Total core flow as a function of THERMAL POWER within Region II of Figure 3.4.1-1. 3.4 #17	C ~~B.~~1 Determine APRM and LPRM neutron flux noise levels.	Once per 8 hours AND 30 minutes after an increase of ≥ 5% RTP
D ~~C.~~ Total core flow as a function of THERMAL POWER within Region II of Figure 3.4.1-1. 3.4 #17 AND APRM or LPRM neutron flux noise level > 3 times established baseline noise level.	D ~~C.~~1 ~~Initiate action to~~ restore APRM and LPRM neutron flux noise level to ≤ 3 times established baseline noise levels.	~~Immediately~~ 2 hours 3.4 #17
B. THERMAL POWER > 70% RTP during single recirculation loop operation.	C.1 Reduce THERMAL POWER to ≤ 70% RTP	1 hour

INSERT 1B (continued)

D. E Total core flow as a function of THERMAL POWER within Region III of figure 3.4.1-1.	D.1 E ~~Initiate action to~~ restore total core flow as a function of THERMAL POWER to within Region I or II of Figure 3.4.1-1.	~~Immediately~~ 4 hours
E. F No recirculation loops in operation.	E.1 F ~~Initiate action to~~ reduce THERMAL POWER to within Region I of Figure 3.4.1-1.	~~Immediately~~ 4 hours
	AND	
	E.2 F Be in MODE 2.	6 hours
	AND	
	E.3 F Be in MODE 3.	12 hours
G. Required limit and setpoint modifications not performed.	G.1 Declare associated limit(s) and setpoint(s) not met.	Immediately

3.4 #17

3.4 #17

3.4 #74

LAK 93-14R1

INSERT 2A

SR 3.4.1.2	~~Verify total core flow as a function of THERMAL POWER to be within Region I.~~	24 hours

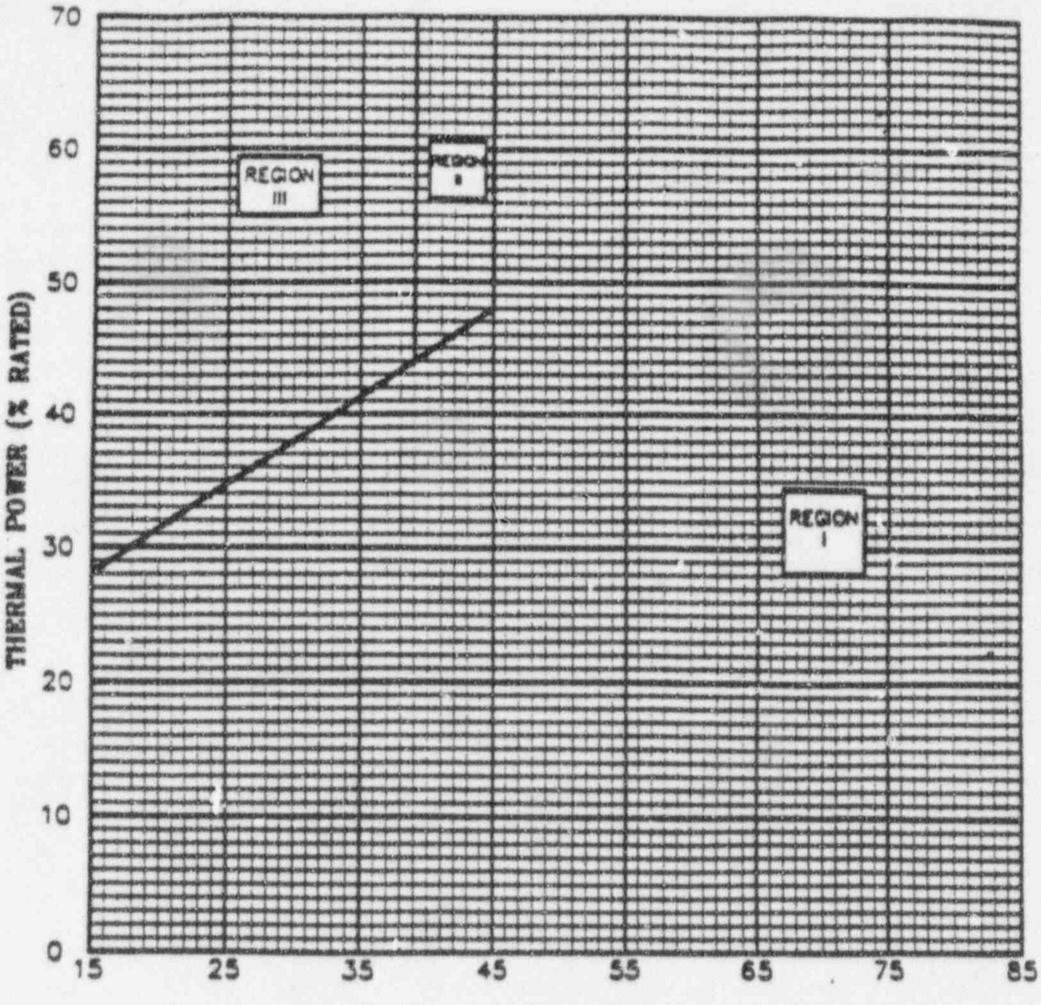
3.4 #17

Verify:

a. Total core flow ≥ 45% rated core flow; or

b. THERMAL POWER and total core flow within Region I of Figure 3.1.1-1,

LAR 95-14R1



~~Bases~~ Figure 3.4.1-1 (page i of 1)
Thermal Power/Core Flow Stability Regions

3.4 #17

INSERT
~~B~~ 3.4-~~6~~2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.4.1 B1	Verify the safety function lift setpoints of the [required] S/RVs are as follows: B1 Number of S/RVs: 7 ~~[8]~~, 5 ~~[6]~~, 4 ~~[6]~~ B1 Setpoint (psig): ≥ 1141.7 and ≤ 1165; ≥ 1156.4 and ≤ 1180; ≥ 1166.2 and ≤ 1190 P5 ~~Following testing, lift settings shall be within ± 1%.~~	[In accordance with the Inservice Testing Program ~~or [18] months]~~ 3.4 #36	B1
SR 3.4.4.2	--------------------NOTE-------------------- Valve actuation may be excluded. -- B1 Verify each [required] relief function S/RV actuates on an actual or simulated automatic initiation signal.	[18] months	B1
SR 3.4.4.3	--------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam ~~dome~~ pressure ~~is~~ ~~≥ [950] psig.~~ C3 and flow are adequate to perform the test. -- B1 Verify each [required] S/RV opens when manually actuated.	[18] months on a STAGGERED TEST BASIS for each valve solenoid	B1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued) C1	B.~~2~~ 1	Verify source of unidentified LEAKAGE increase is not service sensitive type 304, ~~or~~ type 316 austenitic stainless steel or other intergranular stress corrosion cracking susceptible material.	4 hours 3.4 #82
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify RCS unidentified LEAKAGE, ~~and~~ total LEAKAGE, and unidentified LEAKAGE increase are within limits. C5	12 ~~8~~ hours P2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal (RHR) shutdown cooling flowpath ~~are not required to meet the requirements of this LCO~~ when in the shutdown cooling mode of operation.

C1 s e

, or during the transition to or from,

ACTIONS

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

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage from one or more RCS PIVs not within limit. C2 One or more flow paths with	check ------------NOTE------------ Each valve used to satisfy Required Action A.1 and Required Action A.2 shall have been verified to meet SR 3.4.6.1 ~~and be in the reactor coolant pressure boundary [or the high pressure portion of the system]~~. ---------------------------- e	3.4 #70 (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	AND		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

3.4 #71

INSERTS deleted and NUREG words retained

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Leakage Detection Instrumentation

LCO 3.4.7 The following RCS leakage detection instrumentation shall be OPERABLE:

B1

a. Drywell and pedestal P4 floor drain sump monitoring system; ~~[and]~~ S 3.4 #46

b. One channel of either drywell atmospheric particulate or atmospheric gaseous monitoring system; [and

c. Drywell air cooler condensate flow rate monitoring system].

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
P4 A. Drywell or pedestal floor drain sump monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- A.1 Restore drywell and pedestal floor drain sump monitoring system to OPERABLE status. S 3.4 #46	P4 30 days

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

LCO 3.4.8 The specific activity of the reactor coolant shall be ~~limited to:~~

3.4 #56

a. DOSE EQUIVALENT I-131 ~~specific activity~~ ≤ [0.2] μCi/gm. B1

and

b. ~~Gross specific activity ≤ 100/Ē μCi/gm.~~

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

----- NOTE -----
LCO 3.0.4 is not applicable
---------- C1

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor coolant specific activity > [0.2] μCi/gm and ≤ 4.0 μCi/gm DOSE EQUIVALENT I-131. (B1)	A.1	Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND		
	A.2	Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Determine DOSE EQUIVALENT I-131.	Once per 4 hours
OR	AND		
Reactor coolant specific activity > [4.0] μCi/gm DOSE EQUIVALENT I-131. (B1)	B.2.1	Isolate all main steam lines.	12 hours
	OR		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3. AND B.2.2.2 Be in MODE 4.	12 hours 36 hours
C. Reactor coolant specific activity > 100/Ē μCi/gm. 3.4 #56	C.1 Isolate all main steam lines. OR C.2.1 Be in MODE 3. AND C.2.2 Be in MODE 4.	12 hours 12 hours 36 hours C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify reactor coolant gross specific activity is ≤ 100/Ē μCi/gm.	7 days e
SR 3.4.8.~~2~~ 1 3.4 #56 B1	--------------------NOTE-------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ [0.2] μCi/gm.	3.4 #56 ~~31~~ 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.8.3	--------------------NOTE-------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Sample reactor coolant.	184 days
SR 3.4.8.4	Determine Ē from a sample taken in ~~MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~ accordance with SR 3.4.8.3.	~~184 days~~ Once within 92 days of each sample

C5

3.4 #56

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation ~~pump starting~~ temperature requirements shall be maintained within ~~the~~ limits. ~~specified in the PTLR~~ 3.4 #72

loop P2

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODES 1, 2, and 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 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.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. ---------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. --	
3.4 #72	Verify: a. RCS pressure and RCS temperature are within the limits of Figure 3.4.11-1, and b. ~~RCS pressure, RCS temperature, and~~ RCS heatup and cooldown rates are ~~within the limits specified in the PTLR.~~ ≤ 100°F in any one hour period.	30 minutes
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~the PTLR.~~ Figure 3.4.11-1.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
3.4 #72	--------NOTE-------- Only required to be met during control rod withdrawal for the purpose of achieving criticality. --------------------	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.3 B1 3.4 #72	--------------------NOTE-------------------- Only required to be met in MODES 1, 2, 3, and 4 [with reactor steam dome pressure ≥ 25 psig] during recirculation pump start -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 100°F.	3.4 #75 Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4 3.4 #72	--------------------NOTE-------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 50°F.	3.4 #75 Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.5 3.4 #72	--------------------NOTE-------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 70°F.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.6	-------------------NOTE-------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- 3.4 #72 Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR.~~ ≥ 70°F.	30 minutes
SR 3.4.11.7	-------------------NOTE-------------------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. -- 3.4 #72 Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR.~~ ≥ 70°F	12 hours

INSERT 28A — P2

INSERT 28A

SR 3.4.11.8 3.4 #75 during increases in THERMAL POWER or recirculation loop flow 3.4 #72	--------------NOTE------------- Only required to be met in single loop operation with the recirculation loop flow in the operating loop ≤ 50% of rated recirculation loop flow or THERMAL POWER < 30% of RTP. ------------------------------- Verify the difference between the bottom head coolant temperature and the RPV coolant temperature is ~~within the limits specified in the PTLR~~. ≤ 100°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow
SR 3.4.11.9 during increases in THERMAL POWER or Recirculation loop flow 3.4 #75 3.4 #72	--------------NOTE------------- Only required to be met in single loop operation with the recirculation loop flow in the operating loop ≤ 50% of rated recirculation loop flow, or THERMAL POWER < 30% of RTP, and the idle recirculation loop not isolated from the RPV. ------------------------------- Verify the difference between the reactor coolant temperature in the recirculation loop not in operation and the RPV coolant temperature is ~~within the limits Specified in the PTLR~~. ≤ 50°F.	Once within 15 minutes prior to an increase in THERMAL POWER or an increase in loop flow

INSERT FIGURE 3.4.11-1

LAR 93-14R1

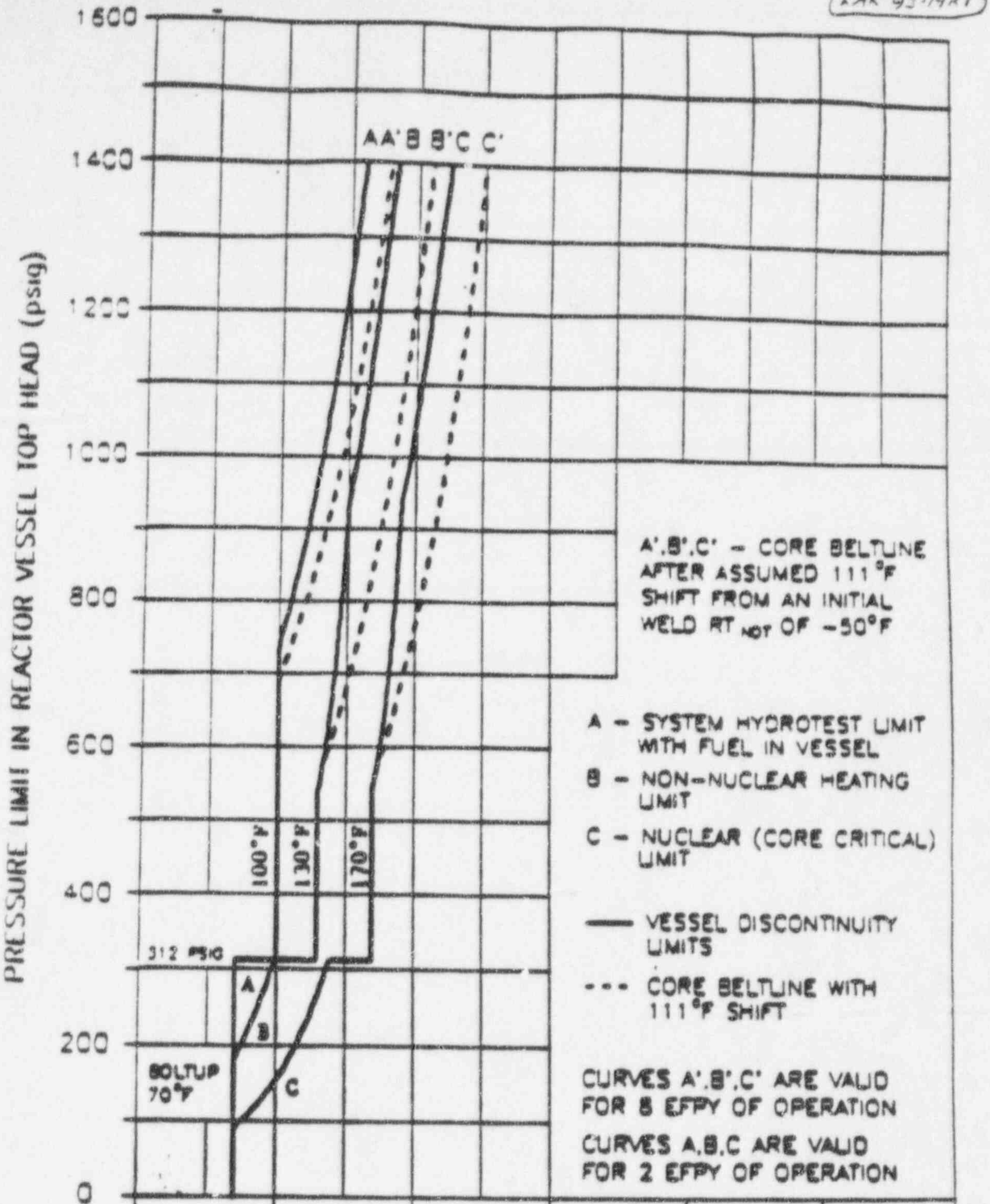


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FIGURE ~~3.4.6.1-2~~ 3.4.11-1 (page 1 of 1)
MINIMUM TEMPERATURE REQUIRED VS ~~REACTOR~~ RCS PRESSURE

INSERT
3.4-28(b)

LAR 95-1401

BASES

BACKGROUND (continued)

The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop can be manually or automatically controlled.

3.4 #72

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. ~~If a LOCA occurs~~ However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, ~~the analysis conservatively assumes the pipe break is~~ with the pipe break assumed to be in the loop with the higher flow. ~~The~~ while the flow coastdown and core response are potentially more severe in this assumed case, (since the intact loop ~~is starting~~ starts at a lower flow rate and the core response is the same as if both loops were operating at the lower flow rate). a small mismatch has been determined to be acceptable based on engineering judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the ~~F~~USAR. P1

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the ~~F~~USAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR ~~setpoints~~ limits (C3) for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

Two recirculation loops are normally (P2) required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. (P2) INSERT B3A (3.4 #17) ~~With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation.~~ With only one recirculation loop in operation, THERMAL POWER must be ≤ 70% RTP, the total core flow limitations identified above must be met, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power—High setpoint (LCO 3.3.1.1) ~~may~~ must (C1) be applied to allow continued operation consistent with the assumptions of Reference 3.

INSERT B3B (P2)

(continued)

LAR 93-14R1

3.4 #17

must be ≥ 45% of rated core flow or total core flow

INSERT B3A

In addition, the total core flow expressed as a function of THERMAL POWER must be in Region I as identified in ~~Bases~~ Figure ~~B~~ 3.4.1-1, "THERMAL POWER/Core Flow Stability Regions." Alternatively, with

3.4 #17

INSERT B3B

The LCO is modified by a Note which allows up to 12 hours before having to put in effect the required modifications to required limits and setpoints after a change in the reactor operating conditions from two recirculation loops operating to single recirculation loop operation. If the required limits and setpoints are not in compliance with the applicable requirements at the end of this period, the associated equipment must be declared inoperable or the limits "not satisfied," and the ACTIONS required by nonconformance with the applicable specifications implemented. This time is provided due to the need to stabilize operation with one recirculation loop, including the procedural steps necessary to limit flow and flow control mode in the operating loop, ~~limit total THERMAL POWER,~~ monitor for excessive APRM and local power range monitor (LPRM) neutron flux noise levels; and the complexity and detail required to fully implement and confirm the required limit and setpoint modifications.

3.4 #17

LAR 93-1481

BASES (continued)

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 both recirculation loops operating but the flows not matched,

With ~~the requirements of the LCO not met~~, the recirculation loops must be restored to operation with matched flows within ~~24~~ 2 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

for more than 2 hours

then

3.4 #17

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

P2

The ~~24~~ 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

~~This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.~~

If the flow mismatch cannot be restored to within limits within 2 hours, one recirculation loop must be shut down.

3.4 #17

(continued)

LAR 93-14x1

3.4 #17 INSERT B5C

BASES

ACTIONS (continued)

C.1, D.1 and E.1

~~B.1, C.1 and D.1~~ P2

P4

INSERT B5A P2

~~With no recirculation loops in operation, or the Required Action and associated Completion Time of Condition A not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < ~~[~~70~~]~~% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < ~~[~~70~~]~~% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop. B1 B1

The mismatch is measured in terms of percent of rated core flow. ~~If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered~~ ~~inoperable~~ not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner. P2

P2 ~~2 hours are provided to attempt to return the loop flows to within limits. After 2 hours,~~

3.4 #17

INSERT B5B P2

(continued)

INSERT B5B

SR 3.4.1.2

This SR ensures the reactor THERMAL POWER and core flows are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. This SR identifies when the conditions requiring interim actions are necessary. The Frequency is based on operating experience and the operators' inherent knowledge of reactor status, including significant changes in THERMAL POWER and core flow.

3.4 #17

~~**SR 3.4.1.3**~~

~~This SR ensures the reactor THERMAL POWER and core flows are within appropriate parameter limits to prevent uncontrolled power oscillations by verifying the APRM and LPRM noise levels. At low flows and high power, the reactor exhibits increased susceptibility to thermal hydraulic instability. This SR identifies when the conditions requiring interim actions are necessary. The Frequency is based on operating experience and the operators' inherent knowledge of reactor status, including significant changes in THERMAL POWER and core flow. The SR is modified by a Note requiring performance of the SR when in the applicable region.~~

INSERT B5C

B.1

3.4 #17

Should a LOCA occur with THERMAL POWER > 70% RTP, during single loop operation the core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 70% RTP.

The 1 hour Completion Time is based on the low probability of an accident occuring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER conditions to be quickly detected.

LAR 93-14r1

3.4 #17 within 2 hours

INSERT B5A

Due to thermal hydraulic stability concerns, operation of the plant is divided into three regions based on THERMAL POWER and core flows. Region III is a power/flow ratio with core flow < 39% of the rated core flow. Region II is a power/flow ratio with core flow ≥ 39% and < 45% of the rated core flow. Deliberate entry into Region III is not permitted, and if it occurs, ~~immediate~~ action is required to exit the region by reducing THERMAL POWER through control rod insertion or by increasing recirculation loop flow by opening the flow control valve. Operation in Region II is also more susceptible to instability than normal operating parameters. However, operation in this region is allowed with the exception that if evidence of instability occurs (i.e., APRM or LPRM neutron flux level is three times the established baseline) then ~~immediate~~ action is required to exit this region.

3.4 #17

A determination of APRM and LPRM neutron flux noise levels every 8 hours provides frequent periodic information relative to established baseline noise levels (see Condition D ~~C~~) that indicate stable steady state operation. A determination of these noise levels within 30 minutes after an increase of ≥ 5% RTP provides a more frequent indication of the stability of operation following any significant potential for change of the thermal hydraulic properties of the system. These Frequencies provide early detection of neutron flux oscillations due to core thermal hydraulic instabilities. ~~Immediate response is required to initiate action to~~ Action must be initiated to restore the plant to a more stable power/flow ratio if such indications of limit cycle neutron flux oscillations are detected.

F.1, F.2, and F.3 ~~E.1, E.2, and E.3~~

With no recirculation loops in operation, the unit is required to be brought to a MODE in which the LCO does not apply. Action must be initiated ~~immediately~~ to reduce THERMAL POWER to be within the limits to assure thermal hydraulic stability concerns are addressed. The plant is then required to be placed in MODE 2 in 6 hours and MODE 3 in 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Times are reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

3.4 #17

The allowed Completion Times are reasonable, based on operating experience, to restore plant parameters to normal in an orderly manner and without challenging plant systems.

LAR 93-14R1

INSERT B5A
(continued)

G.1

3.4 #74

If the required limit or setpoint modifications are not performed within 12 hours after transition from two recirculatrion loop operation to single loop operation, the required limits and setpoints which have not been modified must be immediately declared not met. The Required Actions for the associated limits and instrument channels must then be taken.

LAR 93-14R1

BASES (continued)

REFERENCES

1. ~~U/F~~SAR, Section ~~[5.3.3.4.]~~ 6.3.3 B1
2. ~~U/F~~SAR, Section ~~[5.5.1.5]~~. 5.4.1.4 B1
3. ~~[Plant specific analysis for single loop operation.]~~ USAR, Section 15.0.6. B1

3.4 #17

~~INSERT NEW FIGURE B3.4.1-1~~ P2

INSERT B13A

may need to

under such conditions,

Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships ~~must~~ be re-established each cycle. During the initial weeks of operation ~~in a new cycle,~~ while baselining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

Similarly, initial entry into extened single recirculation loop operation may also require establishment of these relationships.

3.4 #79

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressure transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 2). For the purpose of the analyses, ~~six~~ four of the S/RVs are assumed to operate in the relief mode, and ~~seven~~ five in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the design basis event.

B1 P4

Reference 3 discusses additional events that are expected to actuate the S/RVs. From an overpressure standpoint, ~~these~~ design basis events are bounded by the MSIV closure with flux scram event described above. C5

S/RVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The safety function of ~~seven~~ five S/RVs is required to be OPERABLE in the safety mode, and an additional ~~seven~~ four S/RVs (other than the ~~seven~~ five S/RVs that satisfy the safety function) must be OPERABLE in the relief mode. The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure. In Reference ~~1~~ 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of ~~seven~~ five S/RVs in the safety mode and ~~six~~ four S/RVs in the relief mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded.

P4 3.4 #58

The S/RV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for conditions. The transient evaluations in Reference 3 are based on these setpoints, but also include the additional uncertainties of ~~± 1%~~ -2% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

P3

(continued)

LAR 93-14R1

BASES (continued)

ACTIONS

A.1

C4 LEAKAGE

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the ~~leak~~. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE. However, the total LEAKAGE limit would remain unchanged.

C2

(i.e., reducing the leakage rate such that the current rate is less than the 2 gpm increase in the previous 24 hours; either by isolating the source or other possible methods)

B.1 ~~and B.2~~ C1

increase C2

24 P4

An unidentified LEAKAGE increase of > 2 gpm within a ~~4~~ hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE to within limits is to evaluate RCS type 304 ~~and~~ type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and [redacted] it [redacted] the source of the increased LEAKAGE. ~~This type of piping is very susceptible to IGSCC.~~

increase C2

determine C2 is not

and other intergranular stress corrosion cracking susceptible material

3.4 #82

The 4 hour Completion Time is needed to properly verify the source before the reactor must be shut down.

reduce the LEAKAGE increase or

3.4 #79

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

LAR 93-1421

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

3.4 #80

P5

The RCS LEAKAGE is monitored by a variety of instruments designed to ~~provide alarms when LEAKAGE is indicated and to~~ quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.7, "RCS Leakage Detection Instrumentation." Sump level is ~~and flow rate are~~ typically monitored to determine actual LEAKAGE rates. However, any method may be used to quantify LEAKAGE within the guidelines of Reference 7. In conjunction with alarms and other administrative controls, P2 a 12 ~~an 8~~ hour Frequency for this Surveillance is appropriate for identifying changes in LEAKAGE and for tracking required trends (Ref. 8).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. GEAP-5620, P3 "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
5. C3 NUREG-7~~6~~5/067, P3 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975. B1
6. P1 U~~F~~SAR, Section~~, [5.2.5.5.3]~~.
7. Regulatory Guide 1.45, May 1973. P3
8. Generic Letter 88-01, Supplement 1, P3 "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," February 1992.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 3, valves in the RHR flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the ~~RHR~~ Shutdown Cooling mode of operation. C8 3.4 #79

In MODES 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 has been provided to modify the ACTIONS related to RCS PIV flow paths. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent ~~trains~~ divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. C5 Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for the Condition of RCS PIV leakage limits exceeded provide appropriate compensatory measures for separate affected RCS PIV flow paths. As such, a Note has been provided that allows separate Condition entry for each affected RCS PIV flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable Conditions and Required Actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

A.1 and A.2

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed manual, deactivated automatic, or check valve within 4 hours. ~~Required Action A.1 and Required Action A.2 are~~ C7

C7 C4 INSERT B28A

(continued)

LAR 93-14R1

INSERT B28A

Required Action A.1 and Required Action A.2 are modified by a Note stating that ~~A~~ a check valve may be used for this purpose if leakage past the check valve did not exceed the allowable leakage limit at the last refueling outage, or after the last time the valve was known to have opened, whichever is more recent.

3.4 #70

LAR 93-14?1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of 10 CFR 50, Appendix A (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of rates. The Bases for LCO 3.4.5, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

P2 — LEAKAGE from the RCPB inside the drywell is detected by at least one of ~~two or~~ three independently monitored variables, such as sump level changes and drywell gaseous and particulate radioactivity levels. P4 The primary means of quantifying LEAKAGE in the drywell is the drywell floor (and pedestal) drain sump ~~monitoring system~~. (S) 3.4 #46

The drywell (and pedestal) floor drain sump ~~monitoring system monitors~~ the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, ~~the Closed Cooling Water System~~ (P4: component cooling water, service water,), and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell (and pedestal) floor drain sump(S) ~~has~~ (have; 3.4 #46) transmitters that supply level indications in the main control room. P4

INSERT B31A P4 3.4 #46

~~The floor drain sump level indicators have switches that start and stop the sump pumps when required. A timer starts each time the sump is pumped down to the low level setpoint.~~

(continued)

LAR 93-14R1

BASES

BACKGROUND (continued)

P4

If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit. A second timer starts when the sump pumps start on high level. Should this timer run out before the sump level reaches the low level setpoint, an alarm is sounded in the control room indicating a LEAKAGE rate into the sump in excess of a preset limit. A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room.

P8

atmospheric

The drywell ~~air~~ monitoring systems continuously monitor the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmospheric (P8) particulate and gaseous radioactivity monitoring systems are not capable of quantifying leakage rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

B1

[Condensate from ~~each of~~ the ~~air~~ drywell coolers is routed to the drywell floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This drywell air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.]

APPLICABLE SAFETY ANALYSES

3.4 #80

Identification of the LEAKAGE

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits ~~and providing appropriate alarm of excess LEAKAGE in the control room~~.

~~A control room alarm~~ allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6).

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

3.4 #80

and pedestal (P4) (5) 3.4 #40 (5) are

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, either the flow monitoring or the sump level monitoring portion of the system must be OPERABLE. The other monitoring systems provide ~~early alarms~~ to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

(5)

qualitative indication

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.5. This Applicability is consistent with that for LCO 3.4.5.

ACTIONS

A.1

or pedestal (P4)

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell atmospheric activity monitor [and the drywell air cooler condensate flow rate monitor] will provide indications of changes in leakage. (B1)

or pedestal (P4)

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every ~~8~~ 12 hours (SR 3.4.5.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system

(P6)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1 (continued)

properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.7.2

3.4 #40

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the ~~alarm setpoint and~~ relative accuracy of the instrument~~string~~ation. (P8) The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation. (P8)

SR 3.4.7.3

This SR requires the performance of a CHANNEL CALIBRATION of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument~~string~~ation, (P8) including the instruments located inside ~~containment~~ the drywell (C1). The Frequency of ~~[~~18~~]~~ months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable. (P8)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.

2. Regulatory Guide 1.45, May 1973.

(P1) 3. ~~UF~~SAR, Section ~~[~~5.2.5.2~~]~~. (B1)

4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968. (P3)

5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenetic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975. (P3)

(P1) 6. ~~UF~~SAR, Section ~~[~~5.2.5.5.3~~]~~. (B1)

7. USAR, Table 11.5-1 (P4)

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1).

3.4 #56

This LCO contains ~~both~~ iodine ~~and gross~~ specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. ~~Total specific reactor coolant activity is limited on the basis of the weighted average beta and gamma energy levels in the coolant.~~ The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely. P1

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100.

The limits on specific activity are values from a parametric evaluation of typical site locations. These limits are conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

3.4 #56

The specific iodine activity is limited to ≤ 4 ~~[0.2]~~ μCi/gm (B1) DOSE EQUIVALENT I-131. ~~and the gross specific activity is limited to 100/E μCi/gm.~~ ~~These limits ensure~~ This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated ~~MSIVs closed~~ (C3), such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

C1

INSERT B 39A

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ {0.2} μCi/gm within 48 hours, or if at any time it is > {4.0} μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

B1

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

3.4 #56

(Copy & Repeat)

C4

C6

The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, and C.2.2

When the reactor coolant specific activity is > 100/Ē μCi/gm, all main steam lines must be isolated within 12 hours. ~~Closing the MSIVs eliminates the potential.~~

(continued)

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3.4 #81

INSERT B39A

Required Actions of Condition A

A Note to the ~~ACTIONS~~ excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of a limiting event while exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

BASES

ACTIONS

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

C4

~~radioactivity release path to the environment during an MSLB event.~~

~~Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.~~

3.4 #56

~~The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines without challenging plant systems. Also, the allowed Completion Times for Required Actions C.2.1 and C.2.2 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

SURVEILLANCE REQUIREMENTS

~~SR 3.4.8.1~~

beta and C2

~~This SR requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.~~

C2 beta and

C2 beta and

~~Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a gross fuel failure during this short time frame.~~

3.4 #56

SR 3.4.8.2 1

7

This Surveillance is performed to ensure iodine remains within limit during normal operation. The ~~31~~ 7 day Frequency is adequate to trend changes in the iodine activity level. 2

(continued)

LAR 93-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.8.~~2~~ 1 (continued)

~~considering gross specific activity is monitored every 7 days.~~

3.4 #56

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

C5

~~SR 3.4.8.3 and SR 3.4.8.4~~

~~A radiochemical analysis for Ē determination is required~~ on a periodic basis ~~every 184 days with the plant operating in MODE 1 with equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify plant operation within the gross specific activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. Operating experience has shown that Ē does not change rapidly and the~~ sampling ~~Frequency of 184 days recognizes this.~~

C5

~~This~~ The sampling ~~SR has been modified by a Note that states that sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.~~

C5

INSERT B41A

REFERENCES

1. 10 CFR 100.11~~, 1973.~~ P2
2. UFSAR, Section ~~15.1.40~~ 15.6.4. P1 B1

LAK 93-14x1

INSERT B41A

~~The analysis Frequency is based on a start time coincident with obtaining a reactor coolant sample, and providing time to prepare the sample, conduct the measurement process, and analyze the results as necessary to properly determine Ē.~~

3.4 #56

BASES

LCO (continued)

associated piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

C1

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to be shut down for a period of 2 hours in an 8 hour period~~, provided one subsystem is OPERABLE~~. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests~~, provided the remaining RHR shutdown cooling subsystem is OPERABLE~~. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR system in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

C4

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure ~~above~~ the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures ~~above~~ the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the ~~low pressure~~ RHR shutdown cooling subsystem into operation.

greater than or equal to

3.4 #79

C8

In MODE 3 with reactor steam dome pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR System may be operated in the

(continued)

LAR 93-14R1

BASES

APPLICABILITY (continued)

shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is required to be in operation.

3.4 #79

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown"; LCO 3.9.~~7~~ 8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.~~8~~ 9, "Residual Heat Removal (RHR)—Low Water Level."

C4

ACTIONS

INSERT B44A

C7

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

C2 except as permitted by LCO NOTE 2,

C9

With one ~~or both~~ required RHR shutdown cooling subsystems inoperable for decay heat removal, the inoperable subsystem~~(s)~~ must be restored to OPERABLE status without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability.

C5

~~With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced.~~ Therefore an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

3.4 #79

STET

3.4 #79

C8

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or ~~contributing to~~ the alternate method capability. Alternate methods that can be used include (but are not limited to) the ~~Spent Fuel Pool Cooling~~ Control Rod Drive System or the Reactor Water Cleanup System.

P1

(continued)

LAR 93-14R1

BASES

ACTIONS

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

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

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

C2 1

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by ~~the~~ LCO Note, reactor coolant circulation by the RHR shutdown cooling subsystem or one recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature ~~and pressure~~. C6 The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

C10 or recirculation pump 3.4 #79

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling system), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is ~~deemed~~ sub appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications

(continued)

BASES

LCO (continued)

piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

C1

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to be shut down for a period of 2 hours in an 8 hour period~~, provided one subsystem is OPERABLE~~. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for ~~the~~ performance of surveillance tests~~, provided the other loop is OPERABLE and in operation~~. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy. C4

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure ~~above~~ greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures ~~above~~ the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the ~~low pressure~~ RHR shutdown cooling subsystem into operation.

3.4 #79

C8

In MODE 4, the RHR System may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 200°F. Otherwise, a recirculation pump is required to be in operation.

3.4 #79

(continued)

BASES

3.4 #68

APPLICABILITY (continued)

9 10 The requirements for decay heat removal in MODE 3 below the cut in permissive pressure and in MODE 5 are discussed in LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level." 8 9

C6

ACTIONS

INSERT B49A

C5

A.1

C3 except as permitted by LCO Note 2

With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

3.4 #79

STET

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or ~~contributing to~~ the alternate method capability. Alternate methods that can be used include (but are not limited to) the [redacted] System or the Reactor Water Cleanup System.

~~C2~~

P2

Control Rod Drive

B.1 and B.2

C3

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by [redacted] LCO Notes 1 and until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figure 3.4.11-1

3.4 #72

~~The PTLR~~ contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing~~, and data for the maximum rate of change of reactor coolant temperature~~. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The elements of this LCO are:

3.4 #72

C3

C1

a. RCS pressure, temperature, and heatup or cooldown rate ~~are within the limits specified in the PTLR~~ during RCS heatup, cooldown, and inservice leak and hydrostatic testing.

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is within the limit ~~of the PTLR~~ during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow.

C3

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel meets the limit ~~of the PTLR~~ during Recirculation pump startup.

Repeat

C3

d. RCS pressure and temperature are within the criticality limits ~~specified in the PTLR~~, prior to achieving criticality.

e. The reactor vessel flange and the head flange temperatures are within ~~the~~ limits ~~of the PTLR~~ when tensioning the reactor vessel head bolting studs. ~~are tensioned.~~

C3

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification that operation is within ~~PTLR~~ the limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction of minor deviations.

3.4 #72

STET Deleted INFO

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR has been modified by a Note that requires this Surveillance to be met only during control rod withdrawal for the purpose of achieving criticality.

SR 3.4.11.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

3.4 #72

SR 3.4.11.3 and SR 3.4.11.4

Differential temperatures within the applicable ~~[illegible]~~ limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

3.4 #75

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

during recirculation pump start. In addition, SR 3.4.11.3 is only required to be met when

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

B1

and SR 3.4.11.4 have

SR 3.4.11.3 ~~has~~ been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 ~~[with reactor steam dome pressure ≥ 25 psig]~~. In MODE 5, the overall stress on limiting components is lower; therefore, ΔT limits are not required.

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7 (continued)

during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperatures are within ~~the~~ limits ~~specified in the PTLR~~.

3.4 #72

P2

INSERT B58A

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982. (P3)
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. NEDO-21778-A, "Transient Pressure Rises Affecting Fracture Toughness Requirements for BWRs," December 1978. (P3)
8. (P1) U/SAR, Section ~~15.1.26~~ 15.4.4 (B1).

LAR 93-14 R1

INSERT B58A

SR 3.4.11.8 and SR 3.4.11.9

3.4 #72

Differential temperatures within the applicable ~~PTLR~~ limits ensure that thermal stresses resulting from increases in THERMAL POWER or recirculation loop flow during single recirculation loop operation will not exceed design allowances. Performing the Surveillance within 15 minutes before beginning such an increase in power or flow rate provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the change in operation.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.9 is to compare the temperatures of the operating recirculation loop and the idle loop.

when THERMAL POWER or loop flow is being increased when the above

Plant specific test data has determined that the bottom head is not subject to temperature stratification with natural circulation at power levels as low as 36% of RTP or with any single loop flow rate when the recirculation pump is on high speed operation. Therefore, SR 3.4.11.8 and SR 3.4.11.9 have been modified by a Note that requires the Surveillance to be met only ~~under these~~ conditions. The Note for SR 3.4.11.9 further limits the requirement for this Surveillance to exclude comparison of the idle loop temperature if the idle loop is isolated from the RPV since the water in the loop can not be introduced into the remainder of the reactor coolant system.

3.4 #75

are not met

SR 3.4.11.10

the limits specified in the

3.4 #72

Verification that operation is within PTLR ~~limits~~ is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction of minor deviations.

Surveillance for inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified by a Note that requires this Surveillance to be performed only during inservice leakage and hydrostatic testing.

LAR 93-1481

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14 R1

DISCUSSION OF CHANGES TO NUREG-1434
TS 3.4.8 - RCS SPECIFIC ACTIVITY

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording preferences revised to reflect appropriate plant specific requirements.

PLANT SPECIFIC DIFFERENCE

P.1 The safety analysis report for this station is identified as the Updated Safety Analysis Report and is correctly referred to as the USAR.

P.2 The date on this reference to 10 CFR 100 is deleted since the limits are based on the current regulations.

P.3 This comment number is not used for this station.

CHANGE/IMPROVEMENT TO NUREG STS

C.1 A Note is added to the Required Actions for Condition A to indicate that LCO 3.0.4 is not applicable. Entry into the Applicable Modes should not be restricted since the most likely response to the condition is restoration of compliance within the allowed 48 hours. Further, since the LCO limits assure the dose due to a LOCA would be a small fraction of the 10 CFR 100 limit, operation during the allowed time frame would not represent a significant impact to the health and safety of the public.

This comment number is not used for this station.

C.2 ~~Change proposed to provide clarification that both beta and gamma are measured in this surveillance.~~

C.3 Change proposed to provide consistent wording with the LCO.

3.4 #56

C.4 ~~In order to provide additional clarification regarding the control of radioactive material during a postulated MSLB accident, the second sentence of paragraph is replaced with the last sentence copied from the first paragraph of the B discussion.~~

This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
TS 3.4.8 - RCS SPECIFIC ACTIVITY

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

This comment number is not used for this station

3.4 #50

C.5 ~~The analysis time for completing a determination of $\bar{E}$ is currently interpreted to be 184 days from the last determination. However, this may easily be exceeded if an extended outage were to occur. The current Note * implies that $\bar{E}$ determinations are not required until after a sample is taken. Since $\bar{E}$ determination attempts during non-MODE 1 operation provide indeterminate results, the intent is clearly not a simple calendar type Frequency. Therefore, the proposed sampling Frequency is every 184 days with a Note that allows a delay for sampling when recent operation has not been appropriate to provide a reliable sample. Since sampling is the initial point from which an analysis can be done, a one time Frequency for each sample is also proposed. This Frequency is based on the necessary time to perform the analysis onsite or at an offsite laboratory.~~

C.6 This change provides consistency with the wording of other Bases for similar type Required Actions.

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
TS 3.4.11 - RCS PRESSURE AND TEMPERATURE LIMITS

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording preferences revised to reflect appropriate plant specific requirements.

PLANT SPECIFIC DIFFERENCE

P.1 The safety analysis report for this station is identified as the Updated Safety Analysis Report and is correctly referred to as the USAR.

P.2 Additional Surveillance Requirements are included to incorporate current plant specific verifications of single loop operation limits.

P.3 Additional information is included in the references for ease of identification. This additional information may include the title, revision number and/or date.

P.4 This comment number is not used for this station.

CHANGE/IMPROVEMENT TO NUREG STS

C.1 These changes made to the LCO elements to include when each element applies.

C.2 Editorial information is deleted since it is not required information and it is not consistent with the format and content of other proposed Bases.

3.4 #72 C.3 Changes made to be consistent with changes proposed in Section 5.0.

LAR 93-14R1

RIVER BEND

SECTION 3.5

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.5 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-14 R1

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - SHUTDOWN

<TS 3.5.2>

LIMITING CONDITION FOR OPERATION

3.5 #8 L3

3.5.2 At least two of the following shall be OPERABLE:

LA1

a. The low-pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

b. Low-pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

c. Low-pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

d. Low-pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

e. The high-pressure core spray (HPCS) system with a flow path capable of taking suction from the condensate storage tank or suppression pool, as applicable, when these sources of water are OPERABLE per Specification 3.5.3.b, and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

COND A a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or
COND B suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend
L1 ~~CORE ALTERATIONS, and~~ all operations that have a potential for draining
COND C the reactor vessel. Restore at least one subsystem/ system to OPERABLE status within 4 hours or establish PRIMARY CONTAINMENT INTEGRITY - FUEL
COND D HANDLING within the next 8 hours.

A2 A1

LCO 3.5.2 APPL *The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is opened, and water level is maintained within the limits of Specifications 3.9.8
L2 and 3.9.9.

LAR 93-14R1

EMERGENCY CORE COOLING SYSTEMS

~~3/4.5.3~~ SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

⟨TS 3.5.2
TS 3.6.2.2⟩

~~3.5.3~~ The suppression pool shall be OPERABLE:

MOVED TO 3.6.2.2 A1

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least 137,571 ft^3, equivalent to a level of 19'6".

SR 3.5.2.1
SR 3.5.2.2

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least 94,000 ft^3, equivalent to a level of 13'3", except that the suppression pool level may be less than the limit or may be drained provided that:

LCO 3.5.2
COND B & COND C

1. No operations are performed that have a potential for draining the reactor vessel,

TABLE 1.1-1

2. The reactor mode switch is ~~locked~~ (LA1) in the Shutdown or Refuel position,

3. The condensate storage tank contains (LA2) at least 125,000 available gallons of water, equivalent to a level of 11'1", and

3.3 #11

SR 3.5.2.2

4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel. (LA2)

~~R1~~

LA4

~~c. With two OPERABLE suppression pool pumpback system (SPPS) subsystems each consisting of #:~~

~~1. At least one OPERABLE crescent area sump pump and~~

~~2. An OPERABLE flow path to the suppression pool.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*. Moved to ⟨3.6.2.2⟩ - A1

ACTION:

MOVED TO 3.6.2.2 A1

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression pool water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

L1

LCO 3.5.2
(L3) COND A/B
COND C/D

b. In OPERATIONAL CONDITION 4 or 5* with the suppression pool water level less than the above limit or drained and the above required conditions not satisfied, suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. (LA1) Establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING (A2) within 8 hours. (A3)

LCO 3.5.2 APPL

*The suppression pool is not required to be OPERABLE in OPERATIONAL CONDITION 5 provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is open, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. (L2)

~~#The SPPS is not required to be OPERABLE when the suppression pool is not required to be OPERABLE.~~

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JUL 19 1989

3.5 #11

R1

LA4

EMERGENCY CORE COOLING SYSTEMS

ACTION (Continued)

c. With one SPPS subsystem inoperable, restore the SPPS subsystem to OPERABLE status within 31 days or demonstrate the OPERABILITY of the remaining SPPS subsystem at least once per 31 days by:

1. A functional test of the crescent area sump pump, and

2. Demonstrating that the associated flow path can be aligned to the suppression pool.

d. With both SPPS subsystems inoperable, restore one SPPS subsystem to OPERABLE status within 7 days or:

1. In OPERATIONAL CONDITION 1, 2, or 3 be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL CONDITION 4 or 5* provide at least one alternate pumpback method and demonstrate the OPERABILITY of an alternate method within 24 hours and at least once per 24 hours thereafter, otherwise suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the shutdown position. Establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

MOVED TO SR 3.6.2.2.1 (A1) a. 19'6", at least once per 24 hours, in OPERATIONAL CONDITION 1, 2 and 3.

SR 3.5.2.1 SR 3.5.2.2 b. 13'3", at least once per 12 hours, in OPERATIONAL CONDITION 4 and 5.

LCO 3.5.2 & SR 3.5.2.2 (A5)

4.5.3.2 With the suppression pool level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions* to be satisfied.

3.5 #11

R1

LA4

4.5.3.3 At least once per 92 days, the SPPS shall be demonstrated OPERABLE by:

a. Verifying each crescent area sump pump develops 50gpm, and

b. Verifying the flow path can be aligned to the suppression pool.

*The suppression pool is not required to be OPERABLE in OPERATIONAL CONDITION 5 provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is open, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

LAR 93-14R1

ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.5.2 - ECCS SHUTDOWN

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

3.5 #8

L.3 A Note clarifying the alignment requirements of a LPCI subsystem has been included for proposed SR 3.5.2.4. The note is similar to the existing footnote ## to LCO 3.5.1 except that the proposed note allows operation of one of the RHR subsystems in the shutdown cooling mode during MODE 4 and 5, if necessary, and clarifies that the subsystem is still considered OPERABLE for the LPCI mode. Because manual valve positioning removes the capability of the subsystems to respond automatically, the subsystem would be considered inoperable without this note. Although no specific analysis of this condition has been performed, the allowance provided by the note is acceptable because the return to OPERABILITY entails only the repositioning of valves, either remote or locally, and the energy requiring dissipation in MODE 4 and 5, is considerably less than that at 100% power with normal operating temperature and pressure. Further, because of the low probability of an event requiring an ECCS and the certain need for shutdown cooling, it is considered appropriate to have a subsystem aligned for the decay heat removal.

DISCUSSION OF CHANGES
CTS: 3.5.3 - SUPPRESSION POOL

3.5 #11

RELOCATED SPECIFICATIONS

~~R.2~~ LA.4 The suppression pool pumpback system provides control of post-LOCA leakage from the ECCS in the crescent area of the auxiliary building by returning the leakage to the suppression pool. However, the evaluation (No. 377) summarized in NEDO-31466, Supplement 1, determined that this system is not required to mitigate the consequences of any transient or design basis accident. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and the requirements have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 Movement of the reactor mode switch from the shutdown or refuel position is adequately controlled by the MODES definition table (proposed Table 1.1-1), and the requirement to "lock" the mode switch is adequately controlled by plant procedures.

LA.2 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the USAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications.

LA.3 This comment number is not used for this station.

LC.1 This comment number is not used for this station.

"Specific"

L.1 The requirement for suspension of CORE ALTERATIONS with no ECCS available due to a degraded water source is deleted. Refueling LCOs provide requirements to ensure safe operation during CORE ALTERATIONS including required water level above the RPV flange. The ECCS function provides additional protection for loss of vessel inventory events. However, these events are not initiated by, nor is the response of ECCS hampered by, CORE ALTERATION operations.

(ATR 93-14R1)

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

3.5 #8

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.5.2 - ECCS SHUTDOWN

"L3" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Previous analysis indicates that a LPCI line break may initiate an accident. However, since LPCI is currently analyzed for the requested mode of operation, this change will not affect the probability of such an event. This equipment's role is in mitigating and thereby limiting consequences of analyzed events. The LPCI equipment required for OPERABILITY is only interrupted by valve alignment and is still capable of being manually realigned if the LPCI subsystem is needed to mitigate the consequences of design basis accidents. In addition, the proposed Note is applicable when the reactor is shutdown in MODE 4 or 5. Thus, the reactor heat load is much less than in MODE 1 (the MODE assumed in the accident analysis), and another subsystem of the ECCS are still required to be OPERABLE. These changes are consistent with the philosophy stated in the proposed BWR Standard Technical Specifications. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not reduce a margin of safety because the change has no impact on any safety analysis assumption. The clarifying Note recognizes the conflict in the dual purpose system and allows the decay heat removal function to proceed in MODE 4 and 5. Further, it recognizes that the amount of time to realign the LPCI system from the decay heat removal function has no significant impact on the margin of safety because heat loads under these conditions are far below that assumed in the safety analysis. Because the Note allows decay heat removal to

3.5 #8

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.5.2 - ECCS SHUTDOWN

"L3" CHANGE
(continued)

continue, the movement of the plant towards increased safety conditions and reduced energy levels is unimpeded.

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.5 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. (continued) Required Action and associated Completion Time of Condition E or F not met.	G.2	Reduce reactor steam dome pressure to ≤ ~~[150]~~ 100 psig. B1	36 hours
C3 H. HPCS and low pressure core spray (LPCS) Systems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. 3.5 #21 ~~OR~~ ~~One or more ECCS injection/spray subsystems and two or more ADS valves inoperable.~~ OR HPCS System and one or more ADS valves inoperable. OR Two or more ECCS injection/spray subsystems and one or more ADS valves inoperable.	H.1	Enter LCO 3.0.3.	Immediately

<LCO 3.5.2>

LAR 93-14R1

INSERT 8A

	AND D.3 Initiate action to close one door in each primary containment air lock. ~~except during entry and exit under administrative control.~~	Immediately

3.5 #15

----- NOTE -----
Entry and exit is permissible under administrative control.

LAR 93-14R1

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

operation in the RHR shutdown cooling mode during MODE 3 if necessary.

SR 3.5.1.3

P7 accumulator supply

B1 131

Verification every 31 days that ADS air ~~receiver~~ pressure is ≥ ~~[150]~~ psig assures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The designed pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure (Ref. ~~14~~ 13). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This ~~minimum~~ required pressure of ~~[150]~~ 131 psig is provided by the ~~ADS Instrument Air Supply System.~~ The 31 day Frequency takes into consideration administrative control over operation of the ~~Instrument Air Supply System~~ and alarms for low air pressure.

P5

P2 B1 nonsafety related air supply system (SVV) with safety related backup from the penetration valve leakage control system (LSV).

P2 SVV and LSV systems

SR 3.5.1.4

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50, Appendix K, criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 10).

3.5 #28

P7 sufficient to overcome

with a pump differential pressure

The pump flow rates are verified ~~against a system head~~ that is ~~equivalent to~~ the RPV pressure expected during a LOCA. The ~~total system~~ pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during LOCAs. These values may be established during pre-operational testing. P7 The ~~A 92 day~~ Frequency for this Surveillance is in accordance with the Inservice Testing Program requirements.

(continued)

LAR 93-14R1

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS—Shutdown

BASES

BACKGROUND

A description of the High Pressure Core Spray (HPCS) System, Low Pressure Core Spray (LPCS) System, and low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS—Operating."

3.5 #22

APPLICABLE SAFETY ANALYSES

ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one ECCS injection/spray subsystem is required, post LOCA, to maintain the peak cladding temperature below the allowable limit. It is reasonable to assume, based on engineering judgment, that while in MODES 4 and 5, one ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two ECCS subsystems are required to be OPERABLE in MODES 4 and 5. ~~Two OPERABLE ECCS injection/spray subsystems also ensure adequate inventory makeup in the reactor pressure vessel (RPV) in the event of an inadvertent vessel draindown.~~

The ECCS satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two ECCS injection/spray subsystems are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The LPCS System and each LPCI subsystem consist of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. The HPCS System consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank (CST) to the RPV.

One LPCI subsystem (A or B) may be aligned for decay heat removal in MODE 4 or 5 and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align

(continued)

LAR 93-14R1

BASES

BACKGROUND (continued)

C14

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valves in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge line "keep fill" system is designed to maintain the pump discharge line filled with water.

APPLICABLE SAFETY ANALYSES

3.5 #20

INSERT B20A

P8

The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. ~~The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system is included in the Technical Specifications as required by the NRC Policy Statement.~~

LCO

C3

The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity to maintain RPV inventory during an isolation event.

APPLICABILITY

The RCIC System is required to be OPERABLE in MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the ECCS injection/spray subsystems can provide sufficient flow to the vessel.

ACTIONS

A.1 and A.2

C2

If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCS System is immediately verified to be OPERABLE, the RCIC

(continued)

LAR 93-14R1

INSERT B20A

3.5 #25

Should a design basis control rod drop accident occur, the RCIC System can be used in conjunction with the HPCS System to meet the single failure criteria in mitigating the consequences of the event (Ref. 4). The RCIC System is an Engineered Safety Feature for this event and satisfies Criterion 3 of the NRC Policy Statement.

LAR 93-14R1

INSERT B20A

3.5 #25

Should a design basis control rod drop accident occur, the RCIC System can be used in conjunction with the HPCS System to meet the single failure criteria in mitigating the consequences of the event (Ref. 4). The RCIC System is an Engineered Safety Feature for this event and satisfies Criterion 3 of the NRC Policy Statement.

LAR 93-14R1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.5 (continued)

C3

, "Reactor Core Isolation Cooling (RCIC) System Instrumentation,"

ensures that the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 33.

P3 2. U~~F~~SAR, Section ~~[~~5.4.6.2~~]~~. B1

3. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

P8 4. USAR, Section 5.4.6.1

3.5 #32

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
SECTION 3.5 - ECCS and RCIC

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording revised to reflect appropriate plant specific requirements.

PLANT SPECIFIC DIFFERENCE

P.1 This change provides plant specific equipment terminology.

P.2 This change provides plant specific equipment or analysis description.

P.3 The safety analysis report for this station is identified as the Updated Safety Analysis Report and is correctly referred to as the USAR.

P.4 The Condition and Bases are revised to reflect the use of primary containment rather, than secondary containment, for compensatory protection in this situation. Included with the primary containment boundary is the need for penetration isolation (or isolation capability) and closure of the associated air locks. One closed door provides a boundary which is consistent with the normally required closure of secondary containment access doors.

P.5 Changed reference to provide appropriate plant specific reference. Since this reference has previously been noted. the original reference 13 is deleted, and the original reference 14 is being changed to 13.

P.6 This comment number is not used for this station.

P.7 This change is made to provide consistency with the choice of plant specific bracketed information in the Specification.

3.5 #25 P.8 The RCIC System is considered an Engineering Safety Feature for the Control Rod Drop Accident for this plant.

P.9 This comment number is not used for this station.

P.10 This comment number is not used for this station.

P.11 This comment number is not used for this station.

LAR 93-14R1

RIVER BEND

SECTION 3.6

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.6 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-14R1

RECEIVED

MAR 1 4 1989

SDC

CONTAINMENT SYSTEMS - SHUTDOWN

PRIMARY CONTAINMENT ~~INTEGRITY - FUEL HANDLING~~ A1

LIMITING CONDITION FOR OPERATION

<LCO 3.6.1.10>

3.6.1.10 ~~3.6.1.2~~ PRIMARY CONTAINMENT ~~INTEGRITY - FUEL HANDLING~~ shall be OPERABLE ~~maintained~~. A1

APPLICABILITY: Operational Condition* ~~#~~ A2

ACTION:

Cond A Without PRIMARY CONTAINMENT ~~INTEGRITY - FUEL HANDLING~~, suspend handling of irradiated fuel in the primary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. A1

SURVEILLANCE REQUIREMENTS

3.6.1.10 #6

4.6.1.2 PRIMARY CONTAINMENT ~~INTEGRITY - FUEL HANDLING~~ shall be demonstrated:

a. SR 3.6.1.10.1 Within 24 hours prior to entering (A4) and at least once per 24 hours (L1) during Operational Condition* by verifying that all primary containment penetrations required to be closed during accident conditions are closed by hatches, valves, blind flanges, or deactivated automatic valves secured in position.**

b. By verifying each containment air lock is in compliance with the requirements of Specification 3.6.1.4. A3

SR 3.6.1.10.1 NOTE

LCO 3.6.1.10 APPL

*When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

**Up to twelve vent and drain line pathways may be opened under administrative control for the purposes of surveillance testing provided the total calculated flow rate through the open vent and drain line pathways is less than or equal to 70.2 cfm. LA1

~~#See Special Test Exception 3.10.1~~ A2

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT PURGE SYSTEM ⟨LCO 3.6.1.3⟩ &

LIMITING CONDITION FOR OPERATION ⟨LCO 3.6.4.3⟩

LCO 3.6.1.3
~~3.6.1.9~~ The primary containment purge 36-inch supply and exhaust isolation valves shall be OPERABLE and closed except:

3.6.4.3 #16

NOTE 2 to SR 3.6.1.3.1

a. Each 36-inch purge valve may be open (L1) for purge system operation, with such operation limited to 1000 hours* (A3) per 365 days, for reducing airborne activity and for pressure control, and

LCO 3.6.4.3 Req Act A.1 Note 3 to SR 3.6.1.3.1

b. If the SGTS is in the purge flow path, both trains of the SGTS must be OPERABLE, but only one train of SGTS may be operating in the purge flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. ⟹ NOTE 1 to SR 3.6.1.3.1 and SR 3.6.1.3.5

3.6.1.3 #4

ACTION:

INSERT LCO 3.6.1.3 ACTION NOTE 1 (L6)

LCO 3.6.1.3 COND A

a. With a 36-inch primary (L2) containment purge supply or exhaust isolation valve open for more than 1000 hours* (A3) per 365 days, close and/or seal the 36-inch valve or otherwise isolate the penetration within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

COND E

LCO 3.6.4.3 Req Act A.1 & COND B

b. With both SGTS trains in operation in the purge flow path or without both SGTS OPERABLE with one SGTS in the purge flow path, discontinue 36-inch purge system operation and close the open 36-inch valve(s) or otherwise isolate the penetration(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LCO 3.6.1.3 COND D

c. With a primary containment purge supply or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.9.3, restore the inoperable valve(s) to OPERABLE status (L4) within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. (A1)

COND E

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 ~~4.6.1.9.1~~ Each 36 inch primary containment purge supply and exhaust isolation valve shall be verified to be closed at least once per 31 days.

(L2) 4.6.1.9.2 The cumulative time that the 36-inch primary containment purge supply and/or exhaust isolation valves have been open during the past 365 days shall be determined at least once per 7 days.

*Limit of 2000 hours per 365 days from initial fuel loading until 3 months after the first refueling outage. (A3)

LAR 93-1421

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.6.4.2.3 ~~b~~. At least once per 18 months###, during COLD SHUTDOWN or REFUELING, by verifying that, on a containment isolation test signal, each isolation damper actuates to its isolation position. (A5) (L3) secondary containment automatic (A4)

SR 3.6.4.2.2 ~~c~~. By verifying the isolation time to be within its limit when tested ~~pursuant to~~ Specification 4.0.5. once per 92 days (A9) 3.6.4.2 # 13

(A5)
###The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

LAR 93-14R1

MOVED TO TS 5.7.13 A3

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

SR 3.6.4.3.3

d. At least once per 18 months by:

actual or L2

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

A3

MOVED TO TS 5.7.13

LA1

*a) LOCA, and

b) Annulus ventilation exhaust high radiation signal. R1

2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while the filter train is operating at a flow rate of 12,500 cfm ± 10%.

3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

LA1

a) Manual initiation from the control room, and

A5 *b) Simulated automatic initiation signal.

3.6.4.3 # 9

A3 SR 3.6.4.3.4 4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

MOVED TO TS 5.7.13

5. Verifying that the heaters dissipate $\geq$ 61 kw when tested in accordance with ANSI N510-1980 at the design supply voltage.

e. Verifying, after each complete or partial replacement of a HEPA filter bank, that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 12,500 cfm ± 10%.

A5

*The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

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ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

DISCUSSION OF CHANGES
CTS: 3.6.1.2 - PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 The details of this footnote have been removed to procedures. The commitments in this footnote are not requirements set forth in the analyses, and are commitments which will be contained in plant procedures.

"Specific"

3.6.1.10 #6

L.1 The frequency of the surveillance has been changed from 24 hours to every 31 days. The new surveillance interval is consistent with that denoted in:

SR 3.6.1.3.1, and SR 3.6.1.3.2 of LCO 3.6.1.3, "PCIVs"
SR 3.6.4.1.2 and SR 3.6.4.1.3 of LCO 3.6.4.1 "Secondary Containment- Operating"
SR 3.6.4.2.1 of LCO 3.6.4.2, "SCIDs"
SR 3.6.4.5.2 and SR 3.6.4.5.3 of LCO 3.6.4.5, "Fuel Building".

Moreover, administrative controls ensure that open vent and drain pathways will: (1) only be opened to support leakage rate testing; (2) not exceed 12 valves; (3) require monitoring opened vent and drain valves, as well as the containment-to-auxiliary building differential pressure every 2 hours; and (4) assure at least one person is assigned to each open penetration.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.6.1.9 - PRIMARY CONTAINMENT PURGE SYSTEM

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.2 The time limitations currently applied to opening the containment purge supply and exhaust valves are proposed to be replaced with specific criteria for opening. The time limits were based on engineering judgement and/or early plant operating experience, and not based on any legal requirement. The new limits on when the valves are permitted to be open as defined in Note 2 to SR 3.6.1.3.1 will ensure appropriate controls, and use of the system will be minimized and limited to safety-related reasons. The operating history indicates that these lines are opened only for the proposed specified reasons and for cumulative periods which are generally significantly less than the allowed cumulative times.

L.3 This comment number is not used for this station.

3.6.1.3 #21

L.4 A new action (ACTION ~~E~~ D) is proposed to be added which would permit continued operation with a containment purge supply and/or exhaust valves with resilient seals having a measured leakage rate in excess of the limits as long as the affected penetration is isolated within 24 hours, this isolation is verified every 31 days, and the leak rate test is performed on the purge valves used to perform the isolation every 92 days. These actions assure that the penetration will not leak in excess of limits should an accident occur while operating, and thus alleviate the need to shutdown the facility. If any of the conditions of ACTION ~~E~~ D can not be performed, ACTION ~~F~~ E requires the plant shutdown that is presently required.

L.5 This comment number is not used for this station.

3.6.1.3 #4

L.6 A note has been added to CTS LCO 3.6.1.9 to allow inoperable purge valves to be reopened under administrative controls. This change is consistent with the CTS allowances for PCIVs (including these valves) under LCO 3.6.4. However, since these valves are also addressed under LCO 3.6.1.9 which does not contain this note, reopening these valves is not currently allowed if they are inoperable under LCO 3.6.1.9. This change is consistent with NUREG-1434 and is acceptable based on the limitation that the valves be under administrative control while open. This ensures that the valves will be reclosed promptly in the event containment isolation is required.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.6.5.3 - SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

ADMINISTRATIVE
(continued)

A.8 A Required Action is included to periodically verify that the isolated penetration remains isolated. This verification will assure that if the penetration were inadvertently re-opened, it would eventually be identified. Since this requirement is currently addressed in CTS 4.6.5.1.b this change is considered administrative.

3.6.4.2 #13

A.9 Specification 4.0.5 refers to the inservice inspection and testing of ASME components. Surveillance requirement 4.6.5.3.c infers that the SCID's being tested are ASME components. However, for RBS, the surveillance requirement applies to non-ASME SCID's. During development of the RBS Technical Specifications, the non-ASME SCID's were associated with the inservice inspection and testing program to ensure that they were being tested under a structured program. There was no intent during the development of the Technical Specifications to require additional ASME testing requirements for these non-ASME SCID's

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 An additional surveillance requirement is included to periodically verify that each secondary containment isolation manual valve and blind flange that is required to be closed is closed. These passive isolation devices have not previously been included in the verification of closure except through the ability of the standby gas treatment system to develop and maintain a vacuum. Therefore, this periodic verification constitutes a more restrictive change.

M.2 The proposed specification will now apply to all types of secondary containment isolation devises not just automatic isolation dampers. Since this is an added scope the change is considered more restrictive.

M.3 This comment number is not used for this station.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 This comment number is not used for this station.

⟨DOC LA.2 moved to this page due to addition of DOC A.9 on page 54⟩ (LAR 93-14R1)

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

LA.2 Any time the OPERABILITY of a system or component has been affected by repair, maintenance or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. Explicit post maintenance Surveillance Requirements have therefore been deleted from the specifications.

"Specific"

L.1 An allowance is proposed for intermittently opening closed secondary containment isolation dampers under administrative control as is allowed in the existing primary containment Technical Specifications. The allowance is presented in proposed Actions Note 1 and in SR 3.6.4.2.1 Note 2. Opening of secondary containment penetrations on a intermittent basis is required for many of the same reasons as primary containment penetrations and the potential impact on consequences is less significant.

L.2 In the event both dampers in a penetration are inoperable, the existing Specification, which requires maintaining one isolation damper OPERABLE, would not be met and an immediate shutdown is required. The proposed actions for the secondary containment penetrations provide 4 hours prior to commencing a required shutdown. This proposed 4 hour period is consistent with the existing time allowed for conditions when the secondary containment is inoperable. The proposed change will provide consistency in actions for these various secondary containment degradations.

L.3 The proposed surveillance for a functional test of each secondary containment isolation damper does not include the restriction on plant conditions that requires the surveillance to be performed during Cold Shutdown or Refueling. Some isolations could be adequately tested in other than Cold Shutdown or Refueling, without jeopardizing safe plant operations. The control of the plant conditions appropriate to perform the test is an issue for procedures and scheduling, and has been determined by the NRC Staff to be unnecessary as a Technical Specification restriction. As indicated in Generic Letter 91-04, allowing this control is consistent with the vast majority of other Technical Specification surveillances that do not dictate plant conditions for the surveillance.

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DISCUSSION OF CHANGES
CTS: 3.6.5.4 - STANDBY GAS TREATMENT SYSTEM

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 Details of the methods for performing this surveillance are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the USAR. Additionally, changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications.

3.6.4.3 #1

LA.2 Details related to the design of the system (in this case the standby gas treatment system is designed as two "independent" systems) are discused in the bases of the proposed Technical Specifications. These details do not provide any useful or necessary information to the operator. Additionally, changes to the system design and function are adequately controlled by the requirements of 10 CFR 50.59.

"Specific"

L.1 This comment number is not used for this station.

L.2 The phrase "actual or," in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying that each subsystem actuates on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. OPERABILITY is adequately demonstrated in either case since the subsystem itself can not discriminate between "actual" or "simulated."

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.6.6.2 - PRIMARY CONTAINMENT/DRYWELL HYDROGEN MIXING

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

"Specific"

L.1 A statement is inserted to indicate that Specification 3.0.4 is not applicable for the condition of one primary containment/drywell hydrogen mixing subsystem inoperable. An OPERABLE primary containment/drywell hydrogen mixing subsystem remains available in this condition, and igniters are also available to backup the system. In addition, the purge system does not impact normal operation of the plant in any way, and hence, would not provide any additional initiators for plant transients during startup or MODE changes. Since probabilities have been determined to be acceptable for a 30 day allowed out-of-service time for one division of hydrogen control function equipment, redundant equipment in this system and the other systems is available to perform the function, and there is no impact on normal plant operations from the unavailability of this specific equipment, the exception is considered to provide no significant impact on safety.

L.2 An additional Action is proposed for the Condition of both primary containment/drywell hydrogen mixing subsystems inoperable. The igniters are also designed to control hydrogen in a post-LOCA environment. However, redundancy for the hydrogen control function would be reduced. Therefore, a period of 7 days is proposed to allow attempts to restore at least one division of the purge system to OPERABLE status before requiring a shutdown. This Action would possibly prevent an unnecessary shutdown and the increased potential for transients associated with each shutdown.

L.3 This comment number is not used for this station.

3.6.3.3
#3

L.4 The time limitations currently applied to opening the hydrogen mixing system inlet or outlet valves are proposed to be replaced with specific criteria for opening. The time limits were based on engineering judgement and/or early plant operating experience, and not based on any legal requirement. The new limits on when the valves are permitted to be open as defined in the Note to SR 3.6.5.3.2 will ensure appropriate controls, and use of the system will be minimized and limited to safety-related reasons. The operating history indicates that these lines are opened only for the proposed specified reasons and for cumulative periods which are generally significantly less than the allowed cumulative times.

LAR 93-14R1

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.6.1.2 - PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING

"L1" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

3.6.1.10 #6

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change would increase the surveillance frequency of the verification of the primary containment penetration flow paths. This increase in frequency is consistent with the frequencies during operation of the plant. The proposed change is offset by administrative controls that ensure open vent and drain pathways will: (1) only be opened to support leakage rate testing; (2) not exceed 12 valves; (3) require monitoring opened vent and drain valves, as well as the containment-to-auxiliary building differential pressure every 2 hours; and (4) assure at least one person is assigned to each open penetration. This provides additional assurance that the conditions required by this surveillances are being maintained. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. Further, since the change impacts only the frequency of verification and does not result in any change in the response of the equipment to an accident, the change does not increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the frequency of verification and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

   The change impacts only the frequency of verification of primary containment penetration flow paths. This change does not involve a significant reduction in the margin of safety because of the additional administrative controls required. This essentially confirms the capability to maintain the building leak tightness.

3.6.1.3
#4

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.6.1.9 - PRIMARY CONTAINMENT PURGE SYSTEM

"L6" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will add a note to allow inoperable purge valves to be reopened under administrative controls. Failure of purge valves to close is not assumed to initiate any accidents. Therefore, this change does not increase the probability of a previously analyzed accident. Because the valves are required to under administrative controls when opened under the proposed note, adequate assurance is provided that the valves will be reclosed promptly in the event containment isolation is required. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue require administrative controls to be in place while the valve(s) are open. This provides adequate assurance that the valves will be reclosed promptly in the event containment isolation is required.

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.6.6.2 - PRIMARY CONTAINMENT/DRYWELL HYDROGEN MIXING

"L4" CHANGE

3.6.3.3 #3

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change would replace quantitative restrictions for opening of the hydrogen mixing system inlet or outlet valves with qualitative restrictions. The hydrogen mixing system is not considered as the initiator for any previously evaluated accidents and, therefore, revising the opening criteria will not significantly increase the probability of any previously evaluated accident. Further, the change maintains the current method of operation and response of the hydrogen mixing valves to an accident. Therefore, the change does not increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the opening criteria for the system and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

   This change impacts only the opening criteria for the hydrogen mixing system. The methodology and limits of the accident analysis are not affected, nor is the system response. Therefore, the change does not involve a significant reduction in the margin of safety.

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.6 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. 3.6.1.1 #11 B1 ~~The maximum allowable leakage rate, L_a, is ≤ [ ]% of primary containment air weight per day at the calculated peak containment pressure, P_a.~~ 0.26	-----NOTE------ SR 3.0.2 is not applicable -------------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
~~SR 3.6.1.1.2~~	~~Verify primary containment structural integrity in accordance with the Primary Containment Tendon Surveillance Program.~~ B1	~~In accordance with the Primary Containment Tendon Surveillance Program~~

The leakage rate acceptance criteria is ≤ 1.0 La. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria is < 0.6 La for the Type B and Type C tests and < 0.75 La for the Type A test.

LAR 93-14R1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.2.1 P3 3.6.1.2 #8+9 P7	------------------NOTES------------------ 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. During MODES 1, 2, and 3, Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. --- Perform required primary containment air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions and the acceptance criteria of SR 3.6.1.1.1. ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate is ≤ [2 scfh] when tested at ≥ P_a.~~ ~~b. For each door, leakage rate is ≤ [2 scfh] when the gap between the door seals is pressurized to ≥ [1.0 P_a].~~	-----NOTE------ SR 3.0.2 is not applicable -------------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions 3.6.1.2 #8+9
B1 SR 3.6.1.2.2	Verify primary containment air lock seal air flask pressure is ≥ ~~[~~90~~]~~ psig.	7 days

(continued)

Verify the combined leakage rate is ≤ 13,500 cc/hr for all required annulus bypass leakage paths when pressurized to ≥ Pa.

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

3.6.1.3 #42

APPLICABILITY: MODES 1, 2, and 3,
When associated isolation instrumentation is required to be OPERABLE ~~per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."~~ per LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.b.

MODES 4 and 5 for RHR shutdown cooling system suction from the reactor vessel isolation valves

ACTIONS

-----NOTES-----

B1 1. Penetration flow paths ~~[except for [ ] inch purge valve penetration flow paths]~~ may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment -Operating," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria, in MODES 1, 2, and 3. C7 P9

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~-----NOTE----- Only applicable to penetration flow paths with two PCIVs. -----~~ C8 One or more penetration flow paths with one PCIV inoperable [except ~~for purge valve or~~ due to ~~secondary containment bypass~~ leakage not within limit]. B1 C10	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. 3.6.1.3 #13 AND	4 hours except for main steam line AND 8 hours for main steam line

(continued)

3.6.1.3 #13

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ----------NOTE---------- Only applicable to penetration flow paths with two PCIVs. ---------------------- One or more penetration flow paths with two PCIVs inoperable [except ~~for~~ due to ~~purge valve~~ leakage not within limit]. C14 R1 C10	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ----------NOTE---------- Only applicable to penetration flow paths with only one PCIV. ---------------------- One or more penetration flow paths with one PCIV inoperable. C8	C.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	[4] hours
	AND		
	C.2	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------------------- Verify the affected penetration flow path is isolated.	Once per 31 days
~~D.~~ C. One or more penetration flow paths with ~~Secondary containment bypass~~ leakage rate not within limit except for purge valve leakage. C10	~~D~~.1 C	Restore leakage rate to within limit.	4 hours 2

(continued)

3.6.1.3 #13

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E.~~ D. One or more penetration flow paths with one or more primary containment purge valves not within purge valve leakage limits.	~~E~~ D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	24 hours
3.6.1.3 #20	<u>AND</u>	
B1	~~E~~ D.2 --------NOTE-------- ~~Valves and blind flanges~~ Isolation devices in high radiation areas may be verified by use of administrative means. --------------------	C11
	Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment <u>AND</u> Prior to entering MODE ~~4~~ 2 or 3 from MODE ~~5~~ 4 if not performed within the previous 92 days for isolation devices inside primary containment C5 3.6.1.3 #20
	<u>AND</u>	
	~~E~~ D.3 Perform SR 3.6.1.3.~~6~~ 5 for the resilient seal purge valves closed to comply with Required Action ~~E~~ D.1.	Once per ~~[~~92~~]~~ days

(continued)

LAR 93-14(R1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.6.1.3.1~~	~~[-----NOTE----- Only required to be met in MODES 1, 2, and 3. ------------]~~ ~~Verify each [ ] inch primary containment purge valve is sealed closed except for one purge valve in a penetration flow path while in Condition E of this LCO.~~	~~31 days~~
SR 3.6.1.3.~~2~~ 1	-----NOTES----- [1. Only required to be met in MODES 1, 2, and 3.] Also, not required to be met during 2. Not required to be met when the ~~[20]~~ 36 inch primary containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, ~~or~~ Surveillances or special testing on the purge system that requires the valves to be open. ~~provided the drywell [purge supply and exhaust] lines are isolated.~~ 3. If one Standby Gas Treatment (SGT) subsystem is in the primary containment purge flow path, both SGT subsystems must be OPERABLE. In addition only one SGT subsystem may be operating in the primary containment purge flow path. ------------ Verify each ~~[20]~~ 36 inch primary containment purge valve is closed.	31 days

(continued)

B1

B1

3.6.1.3 #43

P3

3.6.4.3 #10

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~5~~ 4 (B1) 3.6.1.3 #29	Verify the isolation time of each power operated and each automatic PCIV~~[~~, except MSIVs,~~]~~ is within limits.	~~[~~In accordance with the Inservice Testing Program ~~or 92 days]~~ (B1)
~~[~~SR 3.6.1.3.~~8~~ 5 (B1) 3.6.1.3 #44	------------------NOTES------------------ ~~[1.~~ Only required to be met in MODES 1, 2, and 3.~~]~~ ~~2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ --- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days AND Once within 92 days after opening the valve~~]~~
SR 3.6.1.3.~~7~~ 6 (B1) 3.6.1.3 #29	Verify the isolation time of each MSIV is ≥ ~~[~~3~~]~~ seconds and ≤ ~~[~~5~~]~~ seconds.	~~[~~In accordance with the Inservice Testing Program ~~or 18 months]~~ (B1)

(continued)

3.6.1.3 #36

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
P2 SR 3.6.1.3.~~8~~ 7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal. 3.6.1.3 #29	~~[18]~~ months
INSERT 17A → P18		
P2 SR 3.6.1.3.~~8~~ 9 3.6.1.3 #44 C7 B1	------------------NOTE~~S~~------------------ ~~2.~~ ~~Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~1.~~ Only required to be met in MODES 1, 2, and 3. --- Verify the combined leakage rate for all secondary containment bypass leakage paths equipped with PVLCS is ≤ ~~[ ]~~ 170,000 cc/hr when pressurized to ≥ ~~[psig]~~ Pa.	--- NOTE --- SR 3.0.2 is not applicable ------ ~~[18] months~~ In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
P2 SR 3.6.1.3.~~10~~ 10 3.6.1.3 #29 3.6.1.3 #44	------------------NOTE~~S~~------------------ ~~2.~~ ~~Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ ---	
P18	Verify leakage rate through ~~all four main steam lines~~ the valves served by each division of MS-PLCS is ≤ ~~[100]~~ 150 scfh per division when tested at ≥ ~~[11.0] psig~~ Pa.	----NOTE----- SR 3.0.2 is not applicable ------------ In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions

(continued)

LAR 93-14-1

INSERT 17A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~7~~ 8 (3.6.1.3 #29)	Verify in-leakage rate of ≤ 340 scfh for each of the following valve groups when tested at 11.5 psid for MS-PLCS valves and 33 psid for PVLCS sealed valves. a. Division I MS-PLCS valves and Division I PVLCS valves. b. Division II MS-PLCS valves and Division II PVLCS valves. c. Division I MS-PLCS valves and all first outboard PVLCS valves.	18 months

LAR 93-14[illegible]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.11	-----NOTES----- 1. Only required to be met in MODES 1, 2, and 3. 2. ~~Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50 Appendix J, as modified by approved exemptions.~~ -----	---Note--- SR 3.0.2 is not applicable ------- B1
D1 3.6.1.3 # 44 P15 INSERT 18A	Verify combined leakage rate of [1 gpm times the total number of PCIVs] through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at ≥ 1.1 P_a.	~~[18] months~~ B1 IN accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
~~SR 3.6.1.3.12~~	~~-----NOTE----- Only required to be met in MODES 1, 2, and 3. ----- Verify each [ ] inch primary containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[18] months~~

LAR 93-14R1

INSERT 18A

SURVEILLANCE REQUIREMENTS

3.6.1.3 #29

3.6.1.2 #8 & 9

3.6.1.3 #38

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.~~11~~ 12	~~-----NOTE----- Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ 3.6.1.3 #44	----NOTE---- SR 3.0.2 is not applicable
	Verify the combined leakage rate for all annulus bypass leakage paths is ≤ 13,500 cc/hr required when pressurized to ≥ Pa.	In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
P2	SR 3.6.1.9.2 Verify electrical continuity of each inboard MSIV LCS subsystem heater element circuitry.	31 days
P2	SR 3.6.1.9.3 Perform a system functional test of each ~~MSIV LCS subsystem~~. MS-PLCS ~~division~~ subsystem.	[18] months

3.6.1.9 #1

3.6.1.8 #2

(LAR 93-14R1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.10.1	-----------------NOTE~~S~~----------------- ~~1.~~ Not required to be met for vent and drain line pathways provided ~~[~~the total calculated flow rate through open vent and drain pathways is ≤ 70.2 cfm~~] [the reactor has been subcritical for ≥ 7 days]~~. (B1) ~~2. Not required to be met for pathways capable of being closed by OPERABLE primary containment automatic isolation valves.~~ --- Verify each penetration flow path, required to be closed during accident conditions, is closed ~~[or capable of being closed by OPERABLE primary containment automatic isolation valves]~~. (B1)	~~[31 days]~~ (B1) ~~24 hours~~ 31 days 3.6.1.10 #6

LAR 93-14R1

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners ~~(if permanently installed)~~ P2

LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One primary containment hydrogen recombiner inoperable.	A.1	--------NOTE-------- LCO 3.0.4 is not applicable. -------------------- Restore primary containment hydrogen recombiner to OPERABLE status.	30 days
B. Two primary containment hydrogen recombiners inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour ~~AND~~ ~~Once per 12 hours thereafter~~ C1 e 3.6.3.1 # 6
	AND		
	B.2	Restore one primary containment hydrogen recombiner to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

INSERT 39A

B. Two primary containment and drywell hydrogen igniter divisions inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. **AND** B.2 Restore one primary containment and drywell hydrogen igniter division to OPERABLE status.	1 hour ~~AND~~ ~~Once per 12 hours thereafter~~ e 3.6.3.1 #6 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
P2 B. ----------NOTE---------- Separate Condition entry is allowed for each area. ---------------------- One or more open areas with adjacent ignitors inoperable	B.1 Restore adjacent ignitor(s) to OPERABLE status.	30 days
P2 C. ----------NOTE---------- Separate Condition entry is allowed for each area. ---------------------- One or more enclosed areas with no OPERABLE ignitors.	C.1 Restore one ignitor in the enclosed area to OPERABLE status.	7 days
PA C ~~D~~. Required Action and associated Completion Time not met.	C ~~D~~.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.2.1	~~Verify each ignitor is energized when the associated ignitor division is activated.~~ Energize each primary containment and drywell hydrogen igniter division and perform current versus voltage measurements to verify required igniters in service. ~~of each circuit~~	184 days

C2

(continued)

3.6.3.2 #12

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.3.2.2	-------------------NOTE------------------- Not required to be performed until 92 days after discovery of four or more ignitors in the division inoperable. --		
C2	Verify each ignitor is energized when the associated ignitor division is activated.	92 days	
~~SR 3.6.3.2.3~~	~~Visually examine each accessible ignitor to verify cleanliness.~~	~~[18] months~~	B1
SR 3.6.3.2.~~4~~ 3	Verify ~~sufficient~~ current draw ~~to develop~~ for a ≥ ~~[~~1700~~]~~ °F surface temperature ~~for~~ each required ignitors in ~~high radiation or~~ inaccessible areas, develops B1	~~[~~18~~]~~ months	B1
SR 3.6.3.2.~~5~~ 4	Verify ~~the~~ each required igniter in accessible areas develops a surface temperature of ~~each accessible ignitor not in a high radiation area is~~ ≥ ~~[~~1700~~]~~ °F. B1	~~[~~18~~]~~ months	B1

C9

C2 Energize each primary containment and drywell hydrogen ignitor division and perform current versus voltage measurements to verify required igniters in service.

~~of each~~

3.6.3.2 #12

3.6 CONTAINMENT SYSTEMS

3.6.3.3 Primary Containment/Drywell Hydrogen Mixing ~~[Drywell Purge]~~ System] B1

LCO 3.6.3.3 Two primary containment/drywell hydrogen mixing ~~[drywell purge]~~ subsystems shall be OPERABLE. B1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One primary containment/drywell hydrogen mixing ~~[drywell purge]~~ subsystem inoperable. B1	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore primary containment/drywell hydrogen mixing ~~[drywell purge]~~ subsystem to OPERABLE status. B1	30 days
B. Two primary containment/drywell hydrogen mixing ~~[drywell purge]~~ subsystems inoperable. B1	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour ~~AND Once per ~~Every~~ 12 hours thereafter~~ C1 3.6.3.1 #6
	AND		
	B.2	Restore one primary containment/drywell hydrogen mixing ~~[drywell purge]~~ subsystem to OPERABLE status. B1	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
~~C. [Secondary containment] inoperable during movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs.~~	~~C.1~~	~~---------NOTE--------- LCO 3.0.3 is not applicable. ---------------------~~ ~~Suspend movement of irradiated fuel assemblies in the [primary and secondary containment].~~	~~Immediately~~
	~~AND~~		
	~~C.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~		
	~~C.3~~	~~Initiate action to suspend OPDRVs.~~	~~Immediately~~

(B1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
(B1) [SR 3.6.4.1.1	Verify ~~[secondary containment]~~ shield building annulus, auxiliary building and fuel building vacuum is ≥ ~~[0.25]~~ 3.0, ≥ 0.0, and ≥ 0.0 inch of vacuum water gauge, respectively.	24 hours]
SR 3.6.4.1.2	Verify all [secondary containment] equipment hatches are closed ~~and sealed~~. (B1)	31 days

and sealed.

3.6.4.1 #7

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation ~~Valves~~ Dampers (SCIDs)

LCO 3.6.4.2 Each SCID shall be OPERABLE.

(P2)

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the ~~[primary or secondary containment]~~ fuel building for fuel building isolation. (B1)
~~During CORE ALTERATIONS,~~
~~During operations with a potential for draining the reactor vessel (OPDRVs).~~

ACTIONS

---NOTES---

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIDs. (P2)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one SCID inoperable. (P2) 3.6.4.2 #5	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper, or blind flange. (P2)	8 hours
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE-------- ~~Valves and blind flanges~~ Isolation devices C3 in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	 Once per 31 days
B. C4 ~~--------NOTE--------~~ ~~Only applicable to penetration flow paths with two isolation valves.~~ ~~---------------------~~ One or more penetration flow paths with two SCIVs D inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic damper, closed manual damper or blind flange. P2 3.6.4.2 #5	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 **AND** C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.2.1	---NOTES--- 1. Dampers ~~Valves~~ and blind flanges in high radiation areas may be verified by use of administrative ~~controls~~ means. 2. Not required to be met for SCDs that are open under administrative ~~means~~ controls. ------ Verify each secondary containment damper isolation manual ~~valve~~ and blind flange that is required to be closed during accident conditions is closed.	31 days
SR 3.6.4.2.2	Verify the isolation time of each power operated and each automatic SCD is within limits.	92 days ~~[In accordance with the Inservice Testing Program or 92 days]~~
SR 3.6.4.2.3	Verify each automatic SCD actuates to the isolation position on an actual or simulated automatic isolation signal.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
~~During movement of irradiated fuel assemblies in the [primary or secondary containment],~~
~~During CORE ALTERATIONS,~~
~~During operations with a potential for draining the reactor vessel (OPDRVs).~~
P3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.2 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time ~~of Condition A~~ not met. ~~in MODE 1, 2, or 3.~~ (P3)	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
~~C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs.~~ (P3)	~~------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------~~ ~~C.1 Place OPERABLE SGT subsystem in operation.~~ ~~OR~~	~~Immediately~~

3.6.4.3 #16

~~(continued)~~

A.1 ~~Initiate action to~~ Verify OPERABLE SGT subsystem not operating in the primary containment purge flow path | ~~Immediately~~ 4 hours (P3)

AND

3.6.4.3 #16

3.6 CONTAINMENT SYSTEMS

3.6.5.1 Drywell

LCO 3.6.5.1 The drywell shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell inoperable.	A.1 Restore drywell 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 4.	12 hours 36 hours

INSERT 54A P2

SURVEILLANCE REQUIREMENTS

C1 ≤ 3.6.5.1 #6

SURVEILLANCE		FREQUENCY	
SR 3.6.5.1.~~1~~ 3 P2	Verify bypass leakage is less than or equal to ~~≤ [10]% of~~ the bypass leakage limit. B1	~~[18]~~ months	B1
SR 3.6.5.1.~~2~~ 4 P2	Visually inspect the exposed accessible interior and exterior surfaces of the drywell.	~~[40] months~~ Once prior to performance of each Type A test required by SR 3.6.1.1.1	B1 3.6.5.1 #9

However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit. 3.6.5.1 #6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.2.1	~~-----NOTE----- Only required to be performed once after each closing. ---------~~ (3.6.5.2 # 12) B1 Verify seal leakage rate is ≤ ~~[200]~~ 4.05 scfh when the gap between the door seals is pressurized to ≥ ~~[11.5] psig~~ 3.0 psid.	Once within 72 hours after each drywell air lock door closing
SR 3.6.5.2.2	Verify drywell air lock seal air flask pressure is ≥ ~~[90]~~ 75 psig. B1	7 days
SR 3.6.5.2.3	-----NOTE----- Only required to be performed upon entry into drywell. --------- Verify only one door in the drywell air lock can be opened at a time.	18 months
SR 3.6.5.2.4	-----NOTE----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Prior to performance of the overall test at ≥ 3.0 psid, the airlock shall be pressurized to 19.2 psid. (P4) --------- B1 Verify overall drywell air lock leakage rate is ≤ ~~[200]~~ 11.85 scfh by performing an overall air lock leakage test at ≥ ~~[11.5] psig~~ 3.0 psid.	18 months

(continued)

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<< GLOBAL SPELLOUT DIV(s) → drywell isolation valve[s] >> P9 Drywell Isolation Valves ~~DIVs~~

3.6 CONTAINMENT SYSTEMS

3.6.5.3 Drywell Isolation Valves ~~(DIVs)~~

LCO 3.6.5.3 Each DIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. Penetration flow paths, except for the 24 inch purge valve penetration flow path, (P11) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by DIVs.

INSERT NEW NOTE 4

C2 4. Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when DIV leakage results in exceeding overall drywell bypass leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one DIV inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. (3.6.5.3 #8)	~~4 hours for penetration flow paths ≥ [ ] inches in diameter~~ ~~AND~~ ~~8 hours for penetration flow paths < [ ] inches in diameter~~ (C2)
	AND		
			(continued)

LAR 92-1471

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------NOTE--------- ~~Valves and blind flanges~~ Isolation devices C6 in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days
B. C7 ~~---------NOTE--------- Only applicable to penetration flow paths with two isolation valves. ---------------------~~ One or more penetration flow paths with two DIVs inoperable.	B.1	Isolate the affected penetration flow path 3.6.5.3 #8 by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	~~1 hour for penetration flow paths ≥ [ ] inches AND 4 hours~~ for ~~penetration flow paths < [ ] inches~~ C2
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

(LAR 93-14 R1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
B1 SR 3.6.5.3.1	Verify each ~~[ ]~~ 24 inch drywell purge isolation valve is sealed closed.	31 days
SR 3.6.5.3.2	------------------NOTE------------------ Not required to be met when the primary containment/drywell ~~purge supply or exhaust~~ inlet or outlet valves are open for pressure control, ALARA or air quality considerations for personnel entry, or special testing on the hydrogen mixing system. Also not required to be met during Surveillances that require the valves to be open ~~[provided the [20] inch containment [purge system supply and exhaust] lines are isolated]~~. 3.6.5.3 #12 (B1) -- Verify each ~~[20] inch~~ primary containment/drywell ~~purge~~ hydrogen mixing isolation valve is closed.	31 days
SR 3.6.5.3.3	------------------NOTES------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for DIVs that are open under administrative controls. -- Verify each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed. (C8)	Prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

3.6.1.1 #12

a

free standing steel cylinder with an ellipsoidal dome, secured to a steel lined reinforced concrete mat

P4

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material to within limits. The primary containment consists of ~~a steel lined, reinforced concrete vessel~~, which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

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

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic Containment Isolation System, or (C2)

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks";

(C1) c. All equipment hatches are closed; and

(B1) [d. The ~~pressurized~~ sealing mechanism associated with a penetration is OPERABLE. ~~except as provided in LCO 3.6.1.[ ]~~]

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in

(continued)

BACKGROUND (continued)

conformance with 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

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

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

3.6.1.1 #12 — B1

The maximum allowable 0.26 leakage rate for the primary containment (L_a) is ~~0.437~~ % by weight of the containment and drywell air per 24 hours at the maximum peak containment pressure (P_a) of ~~11.5~~ 7.6 psig (Ref. 4).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

3.6.1.1 #11

Primary containment OPERABILITY is maintained by limiting leakage to ~~within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 3)~~ ≤ 1.0 La, except prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test. At this time, the combined Type B and C leakage must be < 0.6 La, and the overall Type A leakage must be < 0.75 La. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the primary containment air locks are addressed in LCO 3.6.1.2.

(continued)

P5

BASES (continued)

APPLICABILITY

P6

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment leakage limits are ~~is~~ not required to be met ~~OPERABLE~~ in MODES 4 and 5 to prevent leakage of radioactive material from primary containment (refer to LCO 3.6.1.10, "Primary Containment - Shutdown").

ACTIONS

A.1

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

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

P2

3.6.1 #2

9

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions. Failure to meet air lock leakage testing (SR 3.6.1.2.1 and SR 3.6.1.2.4), ~~[secondary containment bypass leakage (SR 3.6.1.3.9),]~~ or resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.5), secondary containment bypass leakage (SR 3.6.1.3.8),

(continued)

(LAR 93-14 R1)

P5 - Operating

BASES

, or annulus bypass leakage (SR 3.6.1.3.12)

SURVEILLANCE REQUIREMENTS

3.6.1.1 #11

SR 3.6.1.1.1 (continued)

3.6.1.3 #24

P2 ~~(SR 3.6.1.3.6)~~, main steam ~~isolation valve~~ positive leakage control system (SR 3.6.1.3.10), ~~or~~ hydrostatically tested valve leakage (SR 3.6.1.3.11), does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions (Ref. 3). The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

3.6.1.1 #11

B1

~~[SR 3.6.1.1.2~~

~~The structural integrity of the primary containment is ensured by the successful completion of the Primary Containment Tendon Surveillance Program and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 5).]~~

REFERENCES

P1 1. U~~F~~SAR, Section ~~[~~6.2~~]~~. B1

P1 2. U~~F~~SAR, Section ~~[~~15.6.5~~]~~. B1

3. 10 CFR 50, Appendix J.

P1 4. U~~F~~SAR, Section ~~[ ]~~ 6.2.6. B1

P3 ~~5. Regulatory Guide 1.35, Revision [1].~~

As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be < 0.6 La for combined Type B and C leakage, and < 0.75 La for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 La. At ≤ 1.0 La the offsite dose consequences are bounded by the assumptions of the safety analysis.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Locks

BASES

BACKGROUND

Two double door primary containment air locks have been built into the primary containment to provide personnel access to the primary containment and to provide primary containment isolation during the process of personnel entry and exit. The air locks are designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

3.6.1.2 #17

Each air lock door has been designed and tested to certify its ability to withstand pressure in excess of the maximum expected pressure following a DBA in primary containment. Each of the doors has inflatable seals that are maintained ≥ [60] psig by the seal air flask and pneumatic system, which is maintained at a pressure ≥ [90] psig. Each door has two seals to ensure they are single failure proof in maintaining the leak tight boundary of primary containment.

B1

Each air lock is nominally a right circular cylinder, 10 ft 2 inches in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air locks are provided with limit switches on both doors in each air lock that provide control room indication of door position. [Additionally, control room indication is provided to alert the operator whenever an air lock interlock mechanism is defeated.] During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions, as allowed by this LCO, the primary containment may be accessed through the air lock when the door interlock mechanism has failed, by manually performing the interlock function.

B1

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a

(continued)

<Primary Containment Air Locks
B 3.6.1.2 Applicability>
(LAR 93-14R1)

INSERT B7A

3.6.1.2 #17

Therefore, maintaining OPERABLE primary containment air locks in MODE 4 or 5 to ensure a control volume is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during fuel movement of irradiated fuel assemblies in the primary containment.

INSERT B7B

This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

(LAR 93-14R1)

C6 Since the overall primary containment leakage rate is only applicable to MODE 1, 2, and 3 operation, the NOTE 2 requirement is imposed only during these MODES.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

P3 when in MODES 1, 2, and 3

3.6.1.2 #8&9

INSERT B11A

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria ~~were established during initial air lock and primary containment OPERABILITY testing~~. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1 (P3 during operation in MODES 1, 2, and 3.). This ensures that air lock leakage is properly accounted for in determining the overall primary containment leakage rate.

SR 3.6.1.2.2

B1

The seal air flask pressure is verified to be at ≥ ~~[90]~~ psig every 7 days to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the air lock, which occurs regularly. The 7 day Frequency has been shown to be acceptable through operating experience and is considered adequate in view of the other indications available to operations personnel that the seal air flask pressure is low.

SR 3.6.1.2.3

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed

(LAR 93-14R1)

INSERT B11A

... (i.e., ≤13,500 cc/hr for the combination of all annulus bypass leakage paths that are required to be meeting leak tightness) ensures that the combined leakage rate of annulus bypass leakage paths is less than the specified leakage rate. This provides assurance in MODES 1, 2, and 3 that the assumptions in the radiological evaluations are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (e.g., leakage through the air lock door with the highest leakage) unless the penetration is isolated by use of (for this Specification) one closed and locked air lock door. The leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation devices (e.g., air lock door). If both air lock doors are closed, the actual leakage rate is the lesser leakage rate of the two barriers (doors). This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits used to evaluate Type A, B and C limits are to be quantified in accordance with Appendix J).

3.6.1.2 #8 #9

During the operational conditions of moving irradiated fuel assemblies in the primary containment, CORE ALTERATIONS, or OPDRVS, the only annulus bypass path leakage required to be met is through the two primary containment airlocks; therefore the entire 13,500 cc/hr limit can be applied to the air locks. In these operational conditions the reactor coolant system is not pressurized and specific primary containment leakage limits are not imposed. However, due to the size of the air lock penetration, leakage limits are imposed to assure an OPERABLE barrier. In these conditions the leakage limits are not related to radiological evaluations, but only reflect engineering judgment of an acceptable barrier.

BASES (continued)

a potential flow path (the RHR shutdown Cooling system suction from the reactor vessel) from lowering reactor vessel level to the top of the fuel.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE ~~and the primary containment purge valves are not required to be sealed closed in MODES 4 and 5~~. Certain valves are required to be OPERABLE, however, to prevent ~~inadvertent reactor vessel draindown and release of radioactive material during a postulated fuel handling accident~~. These valves are those whose associated instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," Function 5.b. (This does not include the valves that isolate the associated instrumentation.)

P5 3.6.1.3 #42 3.3.6.1 #25

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) ~~[except for the [ ] inch primary containment purge valve flow path(s)]~~ to be unisolated intermittently under administrative controls. ~~[The primary containment purge valve exception applies to primary containment purge valves that are not qualified to close under accident conditions.]~~ These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. ~~Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the primary containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by the Note to SR 3.6.3.1.~~

B1 P5

A second Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path.

C18 INSERT B16A

The ACTIONS are modified by ~~a third~~ Notes 3 and 4. These NOTES ~~which~~ ensure appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable

(continued)

(LAR 93-14R1)

ACTIONS C1

A.1 and A.2 (continued) C1

primary

containment , drywell, and steam tunnel and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel," is appropriate because the ~~valves~~ devices are operated under administrative controls and the probability of their misalignment is low. For ~~valves~~ devices inside primary containment , drywell, or steam tunnel, the specified time period of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the ~~valves~~ devices and the existence of other administrative controls ensuring that ~~valve~~ device misalignment is an unlikely possibility.

C11 C16 C1 C11

C8

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two PCIVs. For penetration flow paths with one PCIV, Condition C provides appropriate Required Actions.

C11

Required Action A.2 is modified by a Note that applies to ~~valves and blind flanges~~ Isolation devices located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment ~~of these valves~~, once they have been verified to be in the proper position, is low.

C11

C16 except due to leakage not within limits

B.1

With one or more penetration flow paths with two PCIVs inoperable , either the inoperable PCIVs must be restored to OPERABLE status or the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, ~~and~~ a blind flange , and a check valve with flow through the valve secured. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

3.6.1.3 #13 STET 3.6.1.3 #13 C2

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two PCIVs.

C16

C2

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

C8

restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is low.

C.D.1 , annulus bypass leakage rate, hydrostatic leakage rate, or MSIV leakage rate C10

With the secondary containment bypass leakage rate not within limit, the assumptions of the safety analysis ~~are~~ may not be met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolation penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration and the relative importance of ~~secondary containment bypass leakage~~ to the overall containment function.

B1

3.6.1.3 #47

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

3.6.1.3 #20

In the event one or more primary containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve, closed manual valve, and blind flange]. ~~A~~ If a purge valve with resilient seals is utilized to satisfy Required Action D.1, it must have been demonstrated to meet the leakage requirements of SR 3.6.1.3.15. The specified Completion Time is reasonable, considering that one primary containment purge valve remains closed ~~(refer to the note to SR 3.6.1.3.1)~~, so that a gross breach of primary containment does not exist.

B1

C16

3.6.1.3 #20

(continued)

BASES

ACTIONS

D D D
~~E~~.1, ~~E~~.2, and ~~E~~.3 (continued) C16 isolated

primary

3.6.1.3 #20

C17

B1

B1

In accordance with Required Action ~~E~~ D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that primary containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be ~~in the isolation position~~ isolated should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, ~~through a system walkdown~~, that those isolation devices outside primary containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside primary containment, the time period specified as "prior to entering MODE ~~4~~ 2 or 3 from MODE ~~5~~ 4 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of ~~the inaccessibility of the isolation devices and other~~ administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

C5

a a
For ~~the~~ a primary containment purge valve with a resilient seal that is isolated in accordance with Required Action ~~E~~ D.1, SR 3.6.1.3.~~7~~ must be performed at least once every ~~[ ]~~ 92 days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the primary containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.1.3.~~6~~, 184 days, is ~~based on an NRC initiative, Generic Issue B-20 (Ref. 4)~~. Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per ~~[ ]~~ 92 days was chosen and has been shown acceptable based on operating experience.

C21

P2 E E
~~F~~.1 and ~~F~~.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

(continued)

INSERT B23A

If a purge valve is open in violation of this SR, the valve is considered inoperable. ~~(Condition A applies), and unless~~ If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside the of limits. ~~(Condition D).~~

3.6.1.3 #48

3.6.1.3 #34

C16 (automatic isolation capability would be required by SR 3.6.1.3.4, and SR 3.6.1.3.6 7)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.~~2~~ 1 (continued)

[~~valves closing~~. At other times than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open.]

3.6.1.3 #43

or for

The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for pressure control, ALARA, or air quality considerations for personnel entry, ~~or for~~ Surveillances that require the valves to be open, or special testing on the purge system (e.g., testing of the Containment purge radiation monitors). ~~provided the drywell [purge supply and exhaust] lines are isolated~~. These primary containment purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other ~~primary containment purge valve~~ PCIV requirements. ~~discussed in SR 3.6.1.3.1.~~

B1 P16 P3 C21

3.6.1.3 #27

3.6.1.3 #16 3.6.1.3 #49 P2

SR 3.6.1.3.~~3~~ 2

C1

This SR verifies that each primary containment isolation manual valve and blind flange that is located outside primary containment, drywell, ~~or~~ and steam tunnel, and is required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the primary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification ~~through a system walkdown~~, that those ~~valves~~ devices outside primary containment, drywell, and steam tunnel, and capable of being mispositioned, are in the correct position. Since verification of ~~valve~~ position for ~~valves~~ outside primary containment is relatively easy, the 31 day Frequency was chosen to provide added assurance that the ~~valves~~ devices are in the correct positions.

Note 3 to this SR ensures that both SGT subsystems are not damaged should a primary containment pressurization event occur and that one is always available for its' design function.

C17 C11 C1 C11 C11

~~Two~~ Three Notes are added to this SR. The ~~first~~ second Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA

P4 INSERT B24A P4

(continued)

(LAR 95-14-1)

SURVEILLANCE REQUIREMENTS (P2) SR 3.6.1.3.~~3~~ 2 (continued)

reasons. Therefore, the probability of misalignment of these ~~valves~~ (C11 devices), once they have been verified to be in the proper position, is low. A ~~second~~ (third P4) Note is included to clarify that ~~valves~~ (C11 PCIVs) open under administrative controls are not required to meet the SR during the time the ~~valves~~ are open.

(P2) SR 3.6.1.3.~~4~~ 3

This SR verifies that each primary containment manual isolation valve and blind flange located inside primary containment, drywell, or steam tunnel, and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For ~~valves~~ (C11 devices) inside primary containment (C1 , drywell, or steam tunnel), the Frequency of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is appropriate since these ~~valves and flanges~~ are operated under administrative controls and the probability of their misalignment is low. (C20)

INSERT B25A (P4) (P4 three) ~~Two~~ Notes are added to this SR. The ~~first~~ (second P4) Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since access to these areas is typically restricted during MODES 1, 2, and 3. Therefore, the probability of misalignment of these ~~valves~~ (C11 devices), once they have been verified to be in their proper position, is low. A (P4 third) ~~second~~ Note is included to clarify that ~~valves~~ that are open under administrative controls are not required to meet the SR during the time that the ~~valves~~ (PCIVs C11) are open.

(P2) SR 3.6.1.3.~~5~~ 4

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the

(LAR #7)

(continued)

BASES

SURVEILLANCE REQUIREMENTS P2 SR 3.6.1.3.~~5~~ 4 (continued)

B1 safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program ~~or 92 days~~.

SR 3.6.1.3.~~6~~ 5

4 C21

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 6), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. ~~as part of the NRC Resolution of Generic Issue B-20 (Ref. 4).~~ C21 Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

B1 The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are ~~allowed to be open.~~ not required to meet any specific leakage criteria. C16 ~~A second Note is added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that primary containment purge valve leakage is properly accounted for in determining the overall primary containment leakage rate.~~

3.6.1.3 #44

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

3.6.1.3 #29

SR 3.6.1.3.~~7~~ 6

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. The Frequency of this SR is [in accordance with the Inservice Testing Program ~~or 18 months~~]. e

SR 3.6.1.3.~~8~~ 7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.6 overlaps this SR to provide complete test[illegible] of the safety function. The [18] month Frequency is [illegible] on the need to perform this Surveillance under the condit[illegible]ns that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

B1

B1

P14 INSERT B27A

C21

SR 3.6.1.3.9

This SR ensures that the leakage rate of secondary containment bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference ~~6~~ 4 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum

B1 9 e

(continued)

(LAR 93-14R1)

3.6.1.3 #29

INSERT B27A

SR 3.6.1.3.~~7~~ 8

The use of MS-PLCS as a positive leakage barrier results in in-leakage and gradual pressure buildup within the containment. The total allowable MSIV in-leakage rate does not have radiological consequences. This surveillance ensures that the total allowable air in-leakage rate from the MSIVs and valves served by the PVLCS is limited such that containment pressurization does not exceed 50 percent of the design value in a 30 day period due to these sources.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

3.6.1.3 #29

BASES

SURVEILLANCE REQUIREMENTS

9 ~~8~~ P2

SR 3.6.1.3.9 (continued)

pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The ~~[18] month~~ Frequency is ~~based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

required by 10CFR50, Appendix J (Ref. 4), as modified by approved exemptions, thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

B1

C7 INSERT B28A

2

~~A Note is added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that secondary containment bypass leakage is properly accounted for in determining the overall primary containment leakage rate.~~

10 ~~9~~ P2

SR 3.6.1.3.10

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. Leakage through ~~all four MSIVs~~ must be ≤ ~~[100]~~ scfh when tested at ~~≥ [11.5]~~ psig. The ~~MSIV~~ leakage rate must be verified to be in accordance with the leakage test requirements of Reference ~~6~~, as modified by approved exemptions. ~~A Note is added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate.~~ The Frequency is required by 10 CFR 50, Appendix J (Ref. ~~6~~), as modified by approved exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

150

per division

P18 out of the primary containment

3.6.1.3 #44

the valves sealed by each division of MS-PLCS

P18 B1

Pa

2

C7 INSERT B28A

P18

C21 4

C21 4

11 ~~10~~ P2

3.6.1.3 #29

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of References 2 and 3 are met. ~~Note that dual function valves must pass all applicable SRs, including the Type C leakage rate test (SR 3.6.1.1.1), if appropriate.~~ The combined leakage rates must be demonstrated ~~to be in accordance with the~~ leakage

C16

at the frequency of the

(continued)

(LAR 93-14X1)

INSERT B28A (2 places)

A Note ~~3~~ 2 is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the reactor coolant system is not pressurized and specific primary containment leakage limits are not required.

3.6.1.3 #44

3.6.1.3 #46

3.6.1.3 #24

11

28 C13

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

C21 4 C7

test requirements of Reference 6, as modified by approved exemptions; thus SR 3.0.2 (which allows Frequency extensions) does not apply.

A note is added to ...

B1 ~~This SR is modified by two Notes.~~ ~~Note 1~~ states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. ~~However, their leak tightness under accident conditions is not required~~ in these other MODES or conditions. However, specific leakage limits are not applicable C7

3.6.1.3 #44

~~Note 2 is added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that these valve leakages are properly accounted for in determining the overall primary containment leakage rate.~~

P15 INSERT B29A

B1

~~SR 3.6.1.3.12~~

~~[Reviewer's Note: This SR is only required for those plants with purge valves with resilient seals allowed to be open during [MODE 1, 2, or 3] and having blocking devices on the valves that are not permanently installed.]~~

~~[Verifying that each [ ] inch primary containment purge valve is blocked to restrict opening to ≤ [50%] is required to ensure that the valves can close under DBA conditions within the time limits assumed in the analyses of References 2 and 3.]~~

~~[The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.]~~

(LAR 95-14R1)

INSERT B29A

SR 3.6.1.3.~~11~~ 12

3.6.1.3 #29

This SR ensures that the combined leakage rate of annulus bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference 4 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The Frequency is required by 10 CFR 50, Appendix J (Ref. 4), as modified by approved exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

3.6.1.3 #44

~~A Note is added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that secondary containment bypass leakage is properly accounted for in determining the overall primary containment leakage rate.~~

INSERT B39A

... Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ~~ADS~~ LLS valves divert steam flow upon opening.

3.6.1.6 #2

INSERT B39B

The Frequency of 18 months on a STAGGERED TEST BASIS ensures that each solenoid for each S/RV is alternately tested.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area. The containment unit coolers are not required for mitigating LOCA except in the case of steam bypass.

The equivalent flow path area for bypass leakage has been specified to be 1.0 ft^2. The analysis demonstrates that with primary containment unit cooler operation the primary containment pressure and temperature remains within design limits.

3.6.1.7A #1

The primary containment unit coolers satisfy Criterion 3 of the NRC Policy Statement.

LCO

In the event of a Design Basis Accident (DBA), a minimum of one primary containment unit cooler is required to mitigate potential bypass leakage and maintain primary containment peak pressure below design limits. To ensure that these requirements are met, two primary containment unit coolers must be OPERABLE and capable of rejecting heat to the Standby Service Water System. Therefore, in the event of an accident, at least one unit cooler is OPERABLE assuming the worst case single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization and increased temperatures within the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining primary containment unit coolers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one primary containment unit cooler inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE primary containment unit cooler is adequate to perform the primary containment cooling function. However, the overall

(continued)

BASES

SURVEILLANCE REQUIREMENTS

P2 SR 3.6.1.7A.1 (continued)

rather, it involves verification that those dampers capable of being mispositioned are in the correct position.

The 31 day Frequency of this SR is justified because the dampers are operated under procedural control and because improper positioning would affect only a single unit cooler. This Frequency has been shown to be acceptable based on operating experience.

P2 SR 3.6.1.7A.2

B1 Verifying each unit cooler develops a flow rate ≥ [50,000] cfm ensures overall performance has not degraded during the cycle. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is consistent with that applied to pumps by the Inservice Testing Program.

P2 SR 3.6.1.7A.3

B1 This SR verifies that each primary containment unit cooler actuates upon receipt of an actual or simulated automatic actuation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.6 overlaps this SR to provide complete testing of the safety function. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES B1 P1 1. U ~~F~~SAR, Section ~~[6.2.1.1.3.4]~~.

3.6.1.7A #3 — throughout its emergency operating sequence and that the pressure relief and backdraft damper in the flow path actuates to its correct position.

ACTIONS (continued)

C.1 and C.2

If the inoperable PVLCS subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

B1

intermediate and long term post-LOCA

3.6.1.8 #10

The minimum air supply necessary for PVLCS OPERABILITY varies with the system being supplied with compressed air from the PVLCS accumulators. Due to the support system function of PVLCS for S/RV actuator air, however, the specified minimum pressure of [101] psig is required, which provides sufficient air for [ ] S/RV actuations ~~with the drywell pressure at 30 psig~~. This minimum air pressure alone is sufficient for PVLCS to support the OPERABILITY of these S/RV systems and is verified every 24 hours. The 24 hour Frequency is considered adequate in view of other indications available in the control room, such as alarms, to alert the operator to an abnormal PVLCS air pressure condition.

SR 3.6.1.8.2

B1 A simulated system operation is performed every [18] months to ensure that the PVLCS will function throughout its operating sequence. This includes correct automatic positioning of valves once the system is initiated manually. Proper functioning of the compressor and valves is verified

B1 by this Surveillance. The [18] month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. Operating experience has shown that these components usually pass the

B1 Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BACKGROUND (continued)

3.6.1.10 #4

B1

Move to SR 3.6.1.10.1 Bases, pg. B 3.6-53v

[~~Additionally,~~ administrative controls ensure that open vent and drain pathways will: (1) only be opened to support leakage rate testing; (2) not exceed 12 valves; (3) require monitoring opened vent and drain valves, as well as the containment-to-auxiliary building differential pressure every 2 hours; and (4) assure at least one person is assigned to each open penetration (Ref. 1).]

This Specification ensures that the performance of the primary containment, in the event of a fuel handling accident, inadvertent criticality, or reactor vessel draindown, provides an acceptable leakage barrier to contain fission products, thereby minimizing offsite doses.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it contain the fission products from a fuel handling accident inside the primary containment (Ref.2), to limit doses at the site boundary to within limits. The primary containment [performs no active function in response to this event; however, its] ~~[OPERABILITY in conjunction with the automatic closure of~~ selected OPERABLE containment isolation valves (LCO 3.6.1.3, "Primary Containment Isolation Valves," and LCO 3.3.6.1, "Primary Containment Isolation Instrumentation") ~~, assures a leak tight fission product barrier. Its]~~ leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage rates assumed in safety analyses.

B1

[The fuel handling accident inside the primary containment is assumed to occur only after ≥ 80 hours since the reactor was last critical. The fission product release is, in turn, based on an assumed leakage rate from vent and drain valves with a combined flow rate of 70.2 cfm (based on an assumed 0.367 inch water gauge differential pressure)]. This assumed pressure reflects the fact that the fuel handling accident does not produce elevated containment pressures as is the case for the DBA LOCA. However, as an added conservatism, the analysis assumes a non-mechanistic additional leakage of 0.26% of the containment volume per day.]

B1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1 (continued)

(B1) leakage of radioactive ~~[fluids or]~~ gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, or a blind flange. This SR does not require any testing or valve manipulation. Rather, it involves verification that the required valves are in the correct position. The ~~31 day~~ ~~24 hours~~ 31 day Frequency was chosen to provide added assurance that the valves remain in the correct positions. (P2) (3.6.1.10 #6)

The SR is modified by a Note stating that the SR is not required to be met for vent and drain line pathways provided the total calculated flow rate through open vent and drain pathways is ≤ 70.2 cfm. <INSERT FROM pg. B 3.6-53ii> (3.6.1.10 #4)

REFERENCES

(B1) 1. NRC Safety Evaluation Report for River Bend Technical Specification Amendment #35, dated March 3, 1989 ~~[Perry Technical Specification Amendment #44, dated September 28, 1990]~~.

(P1) (B1) 2. USAR, Section 15.7.6.

BASES

BACKGROUND (continued)

B1 When the hydrogen ignitors are energized they heat up to a surface temperature ≥ [1700]°F. At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The hydrogen ignitors depend on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit. C5 ~~Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches [8.0] volume percent (v/o) and results in [85]% of the hydrogen present being consumed.~~

APPLICABLE SAFETY ANALYSES

The hydrogen ignitors cause hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 3). Burning occurs 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 ignitors are not included for mitigation of a Design Basis Accident (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 loss of coolant accident (LOCA). The hydrogen concentration resulting from a DBA can be maintained less than the flammability limit using the hydrogen recombiners. However, the hydrogen ignitors have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for units with Mark III containment.

The hydrogen ignitors are considered to be risk significant in accordance with the NRC Policy Statement.

LCO

a. Two divisions of primary containment and drywell hydrogen ignitors must be OPERABLE, each ~~with~~

P2 more than 90% of the igniters OPERABLE.

3.6.3.2 #11

(i.e., no more than 5 inoperable)

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 [Secondary Containment] - Operating (B1)

BASES

BACKGROUND

The function of the [secondary containment] is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System, Fuel Building Ventilation System (P4) and closure of certain valves whose lines penetrate the [secondary containment], the [secondary containment] is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment. (B1)

3.6.4.1 #18

The [secondary containment] ~~is a structure that~~ consists of the shield building, auxiliary building, and fuel building, and (P4) completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the [secondary containment] to be designed as a conventional structure, the [secondary containment] requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation ~~Valves~~ Dampers (P4) (SCIDs)," ~~and~~ LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," LCO 3.6.4.4, "Shield Building Annulus Mixing System," and LCO 3.6.4.5, "Fuel Building Ventilation System." (P4) (B1)

INSERT B90A

APPLICABLE SAFETY ANALYSES (P4)

There are ~~three~~ two principal accidents for which credit is taken for [secondary containment] OPERABILITY. These are a LOCA (Ref. 1), ~~a fuel handling accident inside primary containment (Ref. 2)~~, and a fuel handling accident in the ~~auxiliary~~ fuel building (Ref. 3). The [secondary containment] performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, (B1) (B1)

(continued)

INSERT B90A

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

3.6.4.1 #18

a. All Auxiliary Building penetrations, Fuel Building penetrations and Shield Building annulus penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation signal, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable, secured in its closed position, except as provided in LCO 3.6.4.2;

b. All Auxiliary Building, Fuel Building and Shield Building Annulus equipment hatches are closed and sealed.

c. The Standby Gas Treatment System is OPERABLE, except as provided in LCO 3.6.4.3;

d. The Fuel Building Charcoal Filtration System is OPERABLE, except as provided in LCO 3.6.4.6; and

e. At least one door in each access to the Auxiliary Building, Fuel Building and Shield Building Annulus is closed, except for routine entry and exit of personnel and equipment.

BASES

ACTIONS

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

(B1)

movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

(B1)

This SR ensures that the ~~[secondary containment]~~ shield building annulus, auxiliary building, and fuel building boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to [secondary containment] vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal [secondary containment] vacuum condition.

3.6.4.1 #7

In this application the term "sealed" has no connotation of leak tightness.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that [secondary containment] equipment hatches and access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the [secondary containment] will not occur. Maintaining [secondary containment] OPERABILITY requires verifying each door in the access opening is closed, except when the access opening is being used for entry and exit~~; then, at least one door must remain closed~~. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other ~~indications of door and hatch status that are available to the operator~~ controls on secondary containment access openings.

(P5)

(P4)

(B1)

[SR 3.6.4.1.4 and] SR 3.6.4.1.~~5~~ 6 (P4)

(B1)

The SGT System exhausts the ~~[secondary containment]~~ shield building annulus and auxiliary building atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated,

(continued)

B 3.6 CONTAINMENT SYSTEMS

P2

B 3.6.4.2 Secondary Containment Isolation ~~Valves~~ Dampers (SC~~V~~Ds)

addressed by this LCO C6

BASES

BACKGROUND P2 The function of the SC~~V~~Ds, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). P2 Secondary containment isolation within the time limits specified for those isolation ~~valves~~ dampers designed to close automatically ensures that fission products that leak from primary containment following a DBA, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within C5 ~~applicable limits.~~ the secondary containment boundary.

P2 The OPERABILITY requirements for SC~~V~~Ds help ensure that an adequate secondary containment ~~leak tightness~~ boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices are either passive or active (automatic). Manual ~~valves~~ dampers, de-activated automatic ~~valves~~ dampers secured in their closed position ~~(including check valves with flow through the valve secured)~~, and blind flanges are considered passive devices. ~~Check valves and~~ other automatic ~~valves~~ dampers designed to close without operator action following an accident are considered active devices. ~~Isolation barrier(s) for the penetration are discussed in Reference 2.~~

C6 C5 C1 3.6.4.2 #14

P2 Automatic SC~~V~~Ds close on a secondary containment isolation signal to C5 establish a boundary for ~~prevent leakage of~~ untreated radioactive material C5 within ~~from~~ secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of C6 dampers or valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES P2 C5 P3 The SC~~V~~Ds must be OPERABLE to ensure ~~that~~ the secondary containment boundary is established ~~is a leak tight~~ barrier to fission product releases. The principal accidents for which the secondary containment ~~leak tightness~~ is required are a loss of coolant accident (Ref. 1), ~~a fuel handling accident inside primary containment (Ref. 3),~~ and a fuel handling accident in the

(continued)

Primary containment is addressed adequately in LCO 3.6.1.10, "Primary Containment-Shutdown." Moving irradiated fuel assemblies in the fuel building (the only portion of secondary containment in which fuel can be handled) will require only the SCIDs associated with the fuel building to be OPERABLE.

P2 D

BASES

APPLICABILITY (continued)

~~CORE ALTERATIONS, or~~ during movement of irradiated fuel assemblies. Moving irradiated fuel assemblies in the ~~[primary or secondary containment] may also occur in MODES 1, 2, and 3.~~ P2 B1

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the ~~valve~~ isolation device. In this way, the penetration can be rapidly isolated when the need for [secondary containment] isolation is indicated. P2 B1

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. C8 INSERT B97A

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV. D P2

A.1 and A.2

3.6.4.2 #5

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criteria are a closed and de-activated automatic SCIV, a closed manual ~~valve, and~~ damper or SCID or a blind flange. For penetrations isolated in accordance with Required Action A.1, the ~~valve~~ device used to isolate the penetration should be the closest available ~~valve~~ to secondary containment. This Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time. D P2 C6 C6 damper damper device

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

Since these ~~valves~~ SCIDs are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the ~~valves~~ SCIDs are in the correct positions. C6 C6

3.6.4.2 #15

Two Notes have dampers been added to this SR. The first Note applies to ~~valves~~ and blind flanges located in high radiation areas and allows them to be verified by use of administrative means ~~controls~~. Allowing verification by administrative ~~controls~~ means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these ~~valves~~ SCIDs, once they have been verified to be in the proper position, is low. P2 C6

3.6.4.2 #14

A second Note has been included to clarify that SCIVs D that are open under administrative controls are not required to meet the SR during the time the ~~valves~~ SCIDs are open. C6

SR 3.6.4.2.2

C6 SCIDs

Verifying the D isolation time of each power operated and each automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the ~~valve~~ will isolate in a time period less than or equal to that assumed in the safety analyses. The ~~isolation time and~~ Frequency of this SR ~~are [in accordance with the Inservice Testing Program or~~ is 92 days].

3.6.4.2 #13

SR 3.6.4.2.3

Verifying that each automatic SCIV D closes on a {secondary containment} isolation signal is required to prevent leakage of radioactive material from {secondary containment} following a DBA or other accidents. This SR ensures that each automatic SCIV D will actuate to the isolation position on a {secondary containment} isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.5 overlaps this SR to provide complete testing of the safety function. The {18} month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant B1 P2 B1 P2 B1

(continued)

SGT System
B 3.6.4.3

LAR 93-14R1

INSERT B104A

A.1 and A.2

With one SGT subsystem inoperable, action must be taken ~~immediately~~ to verify that the OPERABLE SGT subsystem is not operating in the primary containment purge flowpath.

3.6.4.3
#16

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.3 (continued)

The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.2.5 overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the B1 [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

3.6.4.3 #10

SR 3.6.4.3.4

B1

[This SR requires verification that the SGT filter ~~cooler~~ cooling bypass damper can be opened and the fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.]

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. (P1) U FSAR, Section [6.2.3]. (B1)
3. (P1) U FSAR, Section [15.6.5]. (B1)
4. Regulatory Guide 1.52, Rev. [2]. (B1)

BASES (continued)

LCO

3.6.4.4 #6

Following a DBA, a minimum of one shield building annulus mixing subsystem is required to ~~maintain the annulus at a negative pressure with respect to the environment and to~~ adequately mix gaseous releases for processing by the Standby Gas Treatment System. Meeting the LCO requirements for two operable subsystems ensures operation of at least one shield building annulus mixing subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA LOCA could lead to a fission product release to primary containment that leaks to secondary containment, including the annulus. Therefore, Shield Building Annulus Mixing System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of a DBA LOCA event is reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Shield Building Annulus Mixing System OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one shield building annulus mixing subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE shield building annulus mixing subsystem is adequate to perform the required radioactivity release mixing function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release mixing function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant shield building annulus mixing subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES

BACKGROUND (continued)

B1

3.6.4.7 #6

[The Fuel Building Ventilation System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. ~~Following initiation, both enclosure building recirculation fans and both charcoal filter train fans start. Fuel Building Ventilation System flows are controlled by modulating inlet vanes installed on the charcoal filter train exhaust fans and two position volume control dampers installed in branch ducts to individual regions of the secondary containment.~~]

APPLICABLE SAFETY ANALYSES

The design basis for the Fuel Building Ventilation System is to mitigate the consequences of a fuel handling accident (Ref. 3). For all events analyzed, the Fuel Building Ventilation System is shown to reduce, via filtration and adsorption, the radioactive material released to the environment. Since the system is assumed to filter all releases, with the analysis not accounting for any delay in system startup, at least one subsystem must be in operation while handling irradiated fuel.

The Fuel Building Ventilation System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Following a FHA, a minimum of one Fuel Building Ventilation subsystem is required to maintain the fuel building at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two operable subsystems ensures operation of at least one Fuel Building Ventilation subsystem in the event of a single active failure. Requiring one subsystem to be in operation ensures no releases occur that are not filtered and adsorbed.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.1 Drywell

BASES

BACKGROUND

through the suppression pool.

C1

of the suppression pool

The drywell houses the reactor pressure vessel (RPV), the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS), which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a loss of coolant accident (LOCA) to the ~~suppression pool, where it is condensed.~~ Air forced from the drywell is released into the primary containment. The pressure suppression capability assures that peak LOCA temperature and pressure in the primary containment are within design limits. The drywell also protects accessible areas of the containment from radiation originating in the reactor core and RCS.

P2

To ensure the drywell pressure suppression capability, the drywell bypass leakage must be minimized to prevent overpressurization of the primary containment during the drywell pressurization phase of a LOCA. This requires periodic testing of the drywell bypass leakage, confirmation that the drywell air lock is leak tight, OPERABILITY of the drywell isolation valves, ~~(DIVs), and confirmation that the drywell vacuum relief valves are~~ closed.

The isolation devices for the drywell penetrations are a part of the drywell barrier. To maintain this barrier:

a. The drywell air lock is OPERABLE except as provided in LCO 3.6.5.2, "Drywell Air Lock";

b. The drywell penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic DIV, or

C1

P2

3.6.5.1 #4

   2. closed by a manual valve~~s~~, blind flange~~s~~, or the de-activated automatic valve~~s~~ secured in closed position~~s~~ except as provided in ~~LCO~~ 3.6.5.3, "Drywell Isolation Valves (DIVs)" ~~; and~~ ; and

c. All drywell equipment hatches are closed and sealed.

(continued)

BASES

BACKGROUND (continued)

P2

~~c. The Drywell Vacuum Relief System is OPERABLE except as provided in LCO 3.6.5.6, "Drywell Vacuum Relief System."~~

This Specification is intended to ensure that the performance of the drywell in the event of a DBA meets the assumptions used in the safety analyses (Ref. 1).

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 1. The safety analyses assume that for a high energy line break inside the drywell, the steam is directed to the suppression pool through the horizontal vents where it is condensed. Maintaining the pressure suppression capability assures that safety analyses remain valid and that the peak LOCA temperature and pressure in the primary containment are within design limits.

The drywell satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

3.6.5.1 #6

Maintaining the drywell OPERABLE is required to ensure that the pressure suppression design functions assumed in the safety analyses are met. The drywell is OPERABLE if the bypass leakage is within limits ~~and~~ the drywell structural integrity is intact ~~.~~ and

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the drywell is not required to be OPERABLE in MODES 4 and 5.

ACTIONS

A.1

In the event the drywell is inoperable, it must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem

, except prior to the first startup after performing a required drywell bypass leakage test. At this time, the drywell bypass leakage must be ≤ 10% of the drywell bypass leakage limit.

(continued)

BASES

ACTIONS

A.1 (continued)

commensurate with the importance of maintaining the drywell OPERABLE during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring drywell OPERABILITY) occurring during periods when the drywell is inoperable is minimal. Also, the Completion Time is the same as that applied to inoperability of the primary containment in LCO 3.6.1.1, "Primary Containment - Operating." P3

B.1 and B.2

If the drywell cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

P2 - INSERT B110A

3.6.5.1 #10

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.~~1~~ 3 - P2

The analyses in Reference ~~2~~ 1 are based on a maximum drywell bypass leakage. This Surveillance ensures that the actual drywell bypass leakage is ~~≤ [10]% of~~ less than or equal to the acceptable $A/\sqrt{k}$ design value of ~~[~~1.0~~]~~ ft^2 assumed in the safety analysis. B1 The leakage test is performed every [18] months, consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility that a component failure that is not identified by some other drywell or primary containment SR might occur. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[~~18~~]~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. B1

3.6.5.1 #6

As left drywell bypass leakage, prior to the first startup after performing a required drywell bypass leakage test, is required to be ≤10% of the drywell bypass leakage limit. At all other times between required drywell leakage rate tests, the acceptance criteria is based on design $A/\sqrt{k}$. At the design $A/\sqrt{k}$ the containment temperature and pressurization response are bounded by the assumptions of the safety analysis.

SR 3.6.5.1.~~2~~ 4 - P2

The exposed accessible drywell interior and exterior surfaces are inspected to ensure there are no apparent

(continued)

BASES

3.6.5.1 #9

SURVEILLANCE REQUIREMENTS

SR 3.6.5.~~1.2~~ 4 P2 (continued)

physical defects that would prevent the drywell from performing its intended function. This SR ensures that drywell structural integrity is maintained. The ~~[40] month~~ Frequency was chosen so that the interior and exterior surfaces of the drywell can be inspected ~~at every other refueling outage~~ in conjunction with the inspections of the primary containment required by 10 CFR 50, Appendix J (Ref. 2). B1 Due to the passive nature of the drywell structure, the ~~[40] month~~ specified Frequency is sufficient to identify component degradation that may affect drywell structural integrity. B1

REFERENCES

P1 1. U~~F~~SAR, Chapter ~~[~~6~~]~~ and Chapter ~~[~~15~~]~~. B1

BASES

ACTIONS

D.1 and D.2 (continued)

conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2.1

B1

4.05

3.0 psid

This SR requires a test be performed to verify seal leakage of the drywell air lock doors at ~~pressures ≥ [11.5] psig~~. A seal leakage rate limit of ≤ [200] scfh has been established to ensure the integrity of the seals. The Surveillance is only required to be performed once after each closing. The Frequency of 72 hours is based on operating experience. ~~and is considered adequate in view of the other indications available to plant operations personnel that the seal is intact.~~

Within 72 hours

3.6.5.2 #12

SR 3.6.5.2.2

B1

75

Every 7 days the drywell air lock seal air flask pressure is verified to be ≥ [90] psig to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the air lock, which occurs regularly. The 7 day Frequency has been shown to be acceptable, based on operating experience, and is considered adequate in view of the other indications to the plant operations personnel that the seal air flask pressure is low.

SR 3.6.5.2.3

P5

and preventive maintenance

The air lock door interlock is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of the air lock are designed to withstand the maximum expected post accident drywell pressure, closure of either door will support drywell OPERABILITY. Thus, the door interlock feature supports drywell OPERABILITY while the air lock is being used for personnel transit in and out of the drywell. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the

(continued)

LCO (continued) C4 ~~isolation valves and devices are those listed in Reference 2.~~)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the DIVs are not required to be OPERABLE in MODES 4 and 5.

ACTIONS

The ACTIONS are modified by ~~three~~ four (C2) Notes. The first Note allows penetration flow paths, except for the 24 inch purge valve penetration flow paths, (P11) to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the valve. In this way, the penetration can be rapidly isolated when a need for drywell isolation is indicated.

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path. INSERT B122 A - C13

The third Note requires the OPERABILITY of affected systems to be evaluated when a DIV is inoperable. This ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable DIV.

INSERT B122B - C2

<u>A.1 and A.2</u>

3.6.5.3 #17

With one or more penetration flow paths with one DIV inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic ~~DIV~~ valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. In this C10 Condition, the remaining OPERABLE DIV is adequate to perform the isolation function. However, the overall reliability is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.3.1 (continued)

isolation valves ~~that~~ are not qualified to ~~open~~ close under accident conditions. This SR is designed to ensure that a gross breach of drywell is not caused by an inadvertent or spurious drywell purge isolation valve opening. Detailed analysis of these ~~[ ]~~ 24 inch drywell purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to support drywell OPERABILITY. Therefore, these valves are required to be in sealed closed position during MODES 1, 2, and 3. These ~~[ ]~~ 24 inch drywell purge valves that are sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leakage within limits. The Frequency is a ~~result of the NRC resolution of Generic Issue B-24 (Ref. 3) related to~~ based on purge valve use during unit operations.

B1 C14

SR 3.6.5.3.2

This SR ensures that the ~~[20] inch~~ primary containment/drywell ~~purge~~ hydrogen mixing isolation valves are closed as required or, if open, open for an allowable reason. ~~This SR is intended to be used for drywell purge isolation valves that are fully qualified to close under accident conditions; therefore, these valves are allowed to be open~~ for limited periods of time. This SR has been modified by a Note indicating the SR is not required to be met when the primary containment/drywell ~~purge supply or exhaust~~ hydrogen mixing inlet or outlet valves are open for pressure control, ALARA or air quality considerations for personnel entry, or surveillances or special testing of the hydrogen mixing system that require the valve to be open ~~[provided the [20] inch containment [purge system supply and exhaust] lines are isolated]~~. The 31 day Frequency is consistent with the valve requirements discussed under SR 3.6.5.3.1.

B1

3.6.5.3 #18

3.6.5.3 #12

SR 3.6.5.3.3

This SR requires verification that each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that drywell bypass leakage is maintained to a minimum. ~~Since these valves are inside primary containment,~~ Due to the location of these devices,

C10

(continued)

LAR 93-14R2

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
SECTION 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

PLANT SPECIFIC DIFFERENCE
(continued)

P.14 This comment number is not used for this station.

P.15 A surveillance for the annulus bypass leakage has been added consistent with existing requirements.

P.16 Since the drywell purge valves are required to be sealed closed in MODES 1, 2 and 3, no separate restriction is necessary.

P.17 This comment number is not used for this station.

3.6.1.3 #36137

P.18 ~~A~~ surveillances and Bases descriptions for the MS-PLCS ~~has~~ have been added consistent with existing requirements.

P.19 This comment number is not used for this station.

CHANGE/IMPROVEMENT TO NUREG STS

C.1 This change corrects an obvious oversight.

C.2 The improved TS separate primary containment (LCO 3.6.1.1) and PCIVs (LCO 3.6.1.3). The point of separation revolves around Appendix J leakage limits and testing, which are identified as primary containment OPERABILITY. PCIVs form a portion of this boundary (and it is this "boundary that is addressed in LCO 3.6.1.3), however Appendix J leakage limits not met is not a direct impact on PCIV OPERABILITY. Bases discussions of "leak tight" and/or "leakage" related to PCIVs is modified to reflect the more appropriate "boundary" relation.

C.3 This change is proposed to maintain consistency between the Specifications and Bases.

C.4 These changes provide a more detailed and editorially enhanced discussion of the PCIVs design and scope.

C.5 This change provides the appropriate MODE references (as they are also presented in Condition A and SR 3.6.1.3.3.)

C.6 This comment number is not used for this station.

C.7 An exclusion to MODES 1, 2 and 3 is added to Note 4 for clarity. Since LCO 3.6.1.1 is only applicable in MODES 1, 2 and 3, and the overall leakage criteria is not applied outside of these MODES, the exclusion is appropriate.

Similarly, other leakage criteria (SR 3.6.1.3.8 and SR 3.6.1.3.9) have this same exclusion added. This is consistent with the same exclusion already provided in SR 3.6.1.3.10.

LAR 92-14R1

RIVER BEND

SECTION 3.7

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.7 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-14RF

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

MOVED TO TS 5.7.13

A3

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

3. Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

3.7 #11

SR 3.7.3.3

LA.1

SR 3.7.3.4

L5

2. Verifying that on each of the below emergency mode actuation test signals, the subsystem automatically switches to the emergency mode of operation, the isolation valves close within 30 seconds, and the control room is maintained at a positive pressure of ≥ 1/8 inch water gauge relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to 4,000 cfm:

L4

**a) LOCA, and

b) Local air intake radiation monitor - High.

A3

MOVED TO TS 5.7.13

3. Verifying that the heaters dissipate 23 ± 2.3 kw when tested in accordance with ANSI N510-1980, at the design supply voltage.

A7

**The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, schedul- to begin 9-15-87.

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LAR 93-14R1

ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.7.2 - MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

"Specific"

L.1 The Applicability of this specification is revised to exclude MODE 4 & 5 if no activities are being conducted which may lead to a need for control room isolation. The probability and consequences of a design basis accident are significantly reduced due to pressure and temperature limitations in these MODES. However, some activities increase the probability of some accidents and these are retained in the applicability.

L.2 The requirement to perform the system testing on a staggered test basis is deleted. The current requirement delineates unnecessary detail for scheduling of the testing. Since the frequency was not affected, i.e., both current and proposed require monthly testing for each subsystem, and scheduling is not a safety concern as long as both subsystems are not tested simultaneously, this requirement can be deleted with no impact on safety.

L.3 This surveillance requirement is being deleted. In its place an entire new LCO on control room heating, ventilating, and air conditioning (HVAC) is being added in this proposal. Although this 12 hour surveillance requirement is being removed, the new specification better demonstrates the adequacy of the control room HVAC system to remove the necessary heat loads.

L.4 The phrase "actual or simulated" in reference to the automatic initiation signal, has been added to the surveillance requirement for verifying that each subsystem actuates on an automatic initiation signal. This allows satisfactory automatic system initiations for other than surveillance purposes to be used to fulfill the surveillance requirements. OPERABILITY is adequately demonstrated in either case since the system can not discriminate between "actual" or "simulated."

3.7 #11

L.5 The Frequency for demonstrating that the Control Room Ventilation System is capable of pressurizing the control room has been changed from each train every 18 months to one train each 18 months. Since this test is intended to demonstrate the leak tightness of the control room boundary, this test need only be performed on one train each 18 months. The TS will still require verification that each control room ventilation train is capable of automatic initiation every 18 months.

10/17/94

LAR 93-14R1

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

<L4 NSHC found to be missing from original submittal of LAR 93-14> LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.7.2 - CONTROL ROOM EMERGENCY FILTRATION SYSTEM

"L4" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The phrase "actual or simulated" in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated.

   Since the function of the system functional test remains unaffected the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Use of an actual signal instead of the existing requirement which limits use to a simulated signal, will not affect the performance of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "simulated." Therefore, the change does not involve a significant reduction in a margin of safety.

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.7.2 - CONTROL ROOM EMERGENCY FILTRATION SYSTEM

3.7 #11

"L5" CHANGE

Entergy Operations Inc,. has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will decrease the Frequency of demonstrating that each Control Room Ventilation subsystem is capable of pressurizing the control room within the required time limit. control room pressurization time is not assumed to initiate an accident sequences. Therefore, this change does not increase the probability of a previously analyzed accident. Because this test is really a test of the leakage integrity of the control room boundary, it does not need to be performed on each train each refueling outage. This LCO will still require that the automatic initiation capability of each train of control room ventilation be demonstrated each 18 months. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since this test is really a test of the leakage integrity of the control room boundary. Demonstration of this leakage integrity can be accomplished utilizing only one train of control room ventilation each refueling outage. This LCO will still require that the automatic initiation capability of each train of control room ventilation be demonstrated each 18 months.

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.7 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

LAR 92-14R1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
B1	[SR 3.7.1.~~4~~ 3	Operate each ~~[SSW]~~ cooling tower cell fan for ≥ [15] minutes.	31 days]	
F2	SR 3.7.1.~~5~~ 4 B1	~~-------------------NOTE------------------- Isolation of flow to individual components does not render [SSW] System inoperable. --~~ SSW subsystem e 3.7 #26 required B1 Verify each required [SSW] subsystem manual, power operated, and automatic valve in the flow path servicing safety related systems or components, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	
F2	SR 3.7.1.~~6~~ 5	Verify each [SSW] subsystem actuates on an actual or simulated initiation signal.	[18] months	B1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B1 E. Two [CRFA] subsystems inoperable during movement of irradiated fuel assemblies in the B1 [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs.	~~-------NOTE------- LCO 3.0.3 is not applicable. ------------------~~		3.7 #27
	E.1	Suspend movement of irradiated fuel assemblies in the [primary and secondary containment]. B1	Immediately
	AND		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
B1 SR 3.7.3.1 2 P4	Operate each [CRFA] subsystem for [≥ 10 continuous hours with the heaters operating. ~~or (for systems without heaters) ≥ 15 minutes].~~	31 days
B1 SR 3.7.3.2 2 P4	Perform required [CRFA] filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP] B1

(continued)

P3

3.7 PLANT SYSTEMS

3.7.~~5~~ 3 [Control Room Air Conditioning (AC)] System B1

LCO 3.7.~~4~~ 3 Two [control room AC] subsystems shall be OPERABLE. B1

APPLICABILITY: MODES 1, 2, and 3,
B1
During movement of irradiated fuel assemblies in the [primary or secondary containment],
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B1 A. One [control room AC] subsystem inoperable.	A.1 Restore [control room AC] subsystem to OPERABLE status. B1	30 days
C. ~~B.~~ Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C ~~B~~.1 Be in MODE 3. AND C ~~B~~.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.7 #40

B. Two control room AC subsystems inoperable.	B.1 Verify control room area temperature ≤ 104°F. AND B.2 Restore one control room AC subsystem to OPERABLE status.	Once per 4 hours 7 days

P6

3.7 #4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D ~~C~~. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs. (B1)	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	D ~~C~~.1 Place OPERABLE [control room AC]e subsystem in operation. (B1)	Immediately
	OR	
	D ~~C~~.2.1 Suspend movement of irradiated fuel assemblies in the [primary and secondary containment]. (B1)	Immediately
	AND	
	D ~~C~~.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D ~~C~~.2.3 Initiate action to suspend OPDRVs.	Immediately
~~D. Two [control room AC] subsystems inoperable in MODE 1, 2, or 3.~~ (B1)	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~ e

(continued)

3.7 #4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B1 E. Two [control room AC] subsystems inoperable during movement of irradiated fuel assemblies in the B1 [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs. 3.7 #4 Required Action and associated Completion Time of Condition B not met	------------NOTE------------ LCO 3.0.3 is not applicable. -----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the [primary and secondary containment]. B1	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
P3 SR 3.7.3.1	B1 Verify each [control room AC] subsystem has the capability to remove the assumed heat load.	[18] months B1

LAR 93-14R1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
P2 SR 3.7.5.1 2 4	--------------------NOTE-------------------- Not required to be performed until 31 days after any [main steam line not isolated and] SJAE in operation. B1 -- Verify the gross gamma activity rate of the noble gases is ≤ [380] 290 mCi/second [after decay of 30 minutes]. B1	31 days ~~AND~~
SR 3.7.4.1		Once within 4 hours after a ≥ 50% increase in the nominal steady state fission gas release rate after factoring out increases due to changes in THERMAL POWER level

3.7 #16

Copy + repeat

C1 rate

BASES

APPLICABLE SAFETY ANALYSES (continued)

P1 4 is the failure of one of the two standby DGs, which would in turn affect one [SSW] subsystem. The [SSW] flow assumed in the analyses is ~~[7900]~~ 5800 gpm per pump to the ~~heat exchanger~~ RHR P5 (~~F~~USAR, Table ~~[6.2-2]; Ref. 7~~). Reference 2 discusses [SSW] System performance during these conditions. B1

The [SSW] System, together with the [UHS], satisfy Criterion 3 of the NRC Policy Statement.

LCO

The OPERABILITY of subsystem A (Division 1) and subsystem B (Division 2) of the [SSW] System is required to ensure the effective operation of the RHR System in removing heat from the reactor, and the effective operation of other safety related equipment during a DBA or transient. Requiring both subsystems to be OPERABLE ensures that either subsystem A or B will be available to provide adequate capability to meet cooling requirements of the equipment required for safe shutdown in the event of a single failure.

A subsystem is considered OPERABLE when:

P7 a. The associated ~~pump is~~ Pumps are OPERABLE; and

P2 ~~b. The associated [UHS] is OPERABLE; and~~

b ~~c~~. The associated piping, valves, instrumentation, and controls required to perform the safety related function are OPERABLE.

B1 OPERABILITY of the [UHS] is based on a maximum water temperature of ~~[90]~~ 88°F with ~~OPERABILITY of each subsystem requiring~~ P2 a minimum basin water level at or above elevation 111 ft 0 inches ~~[130 ft 3 inches]~~ mean sea level (equivalent to an indicated level of ≥ ~~[7 ft 3 inches]~~ 78%) and four OPERABLE cooling tower ~~fans~~ cells. P7

The isolation of the [SSW] System to components or systems may render those components or systems inoperable, but ~~does~~ may not affect the OPERABILITY of the [SSW] System. B1 3.7 #26

APPLICABILITY

In MODES 1, 2, and 3, the [SSW] System and [UHS] are required to be OPERABLE to support OPERABILITY of the B1

(continued)

LAR 93-M01

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.~~4~~ 3 C7

Operating each cooling tower fan cell for ≥ 15 minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the cooling tower fans occurring between Surveillances. B1

SR 3.7.1.~~5~~ 4 P3

Verifying the correct alignment for each required manual, power operated, and automatic valve in each [SSW] subsystem flow path provides assurance that the proper flow paths will exist for [SSW] operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. 3.7 #26 B1 B1 C1

3.7 #26

~~This SR is modified by a Note indicating that~~ Isolation of the ~~[SSW] System~~ SSW subsystem to components or systems ~~may render those components or systems inoperable, but~~ does not necessarily affect the OPERABILITY of the [SSW] System. As such, when all [SSW] pumps, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the ~~[SSW] System~~ SSW subsystem is still OPERABLE. B1

needs to be evaluated to determine if it

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

(continued)

ACTIONS

D.1 (continued)

the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

3.7 #27

~~The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

During movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs, with two [CRFA] subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. B1

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the [primary and secondary containment] must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended. B1

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 2 P4

This SR verifies that a subsystem in a standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each subsystem once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the

(continued)

BASES (continued)

ACTIONS

A.1

With one [control room AC] subsystem inoperable, the inoperable [control room AC] subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE [control room AC] subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate cooling methods. B1

INSERT B21A

3.7 #4

C C
~~B~~.1 and ~~B~~.2

In MODE 1, 2, or 3, if the control room area temperature cannot be maintained ≤ 104°F or if the inoperable [control room AC] subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. P1

D D D D
~~C~~.1, ~~C~~.2.1, ~~C~~.2.2, and ~~C~~.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE [control room AC] subsystem may be placed immediately in operation. B1

(continued)

LAR 93-14R1

INSERT B21A

once per 4 hours

B.1 and B.2

3.7 #4

If both control room AC subsystems are inoperable, the Control Room AC System may not be capable of performing its intended function. Therefore, the control room area temperature is required to be monitored to ensure that temperature is being maintained low enough that equipment in the control room is not adversely affected. With the control room temperature being maintained within the temperature limit, 7 days is allowed to restore a control room AC subsystem to OPERABLE status. ~~This~~ Completion times ~~is~~ are reasonable considering that the control room temperature is being maintained within limits, the low probability of an event occuring requiring control room isolation, and the availability of alternate cooling methods.

These

BASES

ACTIONS

D D D D

E.1, E.2.1, E.2.2, and E.2.3 (continued)

This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action E.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the [primary and secondary containment] must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended. B1

3.7 #4

~~D.1~~

~~If both [control room AC] subsystems are inoperable in MODE 1, 2, or 3, the [Control Room AC] System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.~~

E.1, E.2, and E.3

~~The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

During movement of irradiated fuel assemblies in the [primary or secondary containment], during CORE ALTERATIONS, or during OPDRVs ~~with two [control room AC] subsystems inoperable~~, action must be taken to immediately suspend activities that present a potential for releasing B1

if the Required Action and associated Completion Time of Condition B is not met,

(continued)

BASES

ACTIONS

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

radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the [primary or secondary containment] must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended. B1

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1 3 P3

3.7 #41

This SR verifies that the heat removal capability of the system is sufficient to remove the ~~assumed~~ heat load assumed in the safety analysis control room. The SR consists of a combination of testing and calculation. The [18] month Frequency is appropriate since significant degradation of the [Control Room AC] System is not expected over this time period. B

REFERENCES

P1 1. FSAR, Section [6.4]. B1

P1 2. FSAR, Section [9.4.1]. B1

LAR 95-14R1

(P2)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 4 2 (P2)

m (P3)

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are ~~Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate.~~ The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable based on operating experience.

C1

release

3.7 #16

INSERT 326A

(B1) This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any [main steam line is not isolated] and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

(P1)

1. ~~F~~SAR, Section [15.7.1]. u (B1)
2. NUREG-0800.
3. 10 CFR 100.

INSERT B26A

SR 3.7.4.1

3.7 #16

This SR requires an isotopic analysis of an offgas sample if the measured release rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88.

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
TS 3.7.4 - CONTROL ROOM AIR CONDITIONING SYSTEM

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording preferences revised to reflect appropriate plant specific requirements.

PLANT SPECIFIC DIFFERENCE

P.1 The safety analysis report for this station is identified as the Updated Safety Analysis Report and is correctly referred to as the USAR.

P.2 This comment number is not used for this station.

P.3 This specification renumbered due to removal of prior non-applicable specification.

P.4 This comment number is not used for this station.

P.5 This change specifically delineates the relationship of the control room air conditioning heating coils to system OPERABILITY.

P.6 Temperature limits are included consistent with current requirements and a more restrictive Completion Time for repair is proposed than is currently required.

3.7 # 4

CHANGE/IMPROVEMENT TO NUREG STS

C.1 Unnecessary details are removed from the Bases Background and Applicability discussions. The USAR references provide sufficient plant specific details regarding design and operation.

LAR 93-14RD

RIVER BEND

SECTION 3.8

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.8 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-1481

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SDC

SURVEILLANCE REQUIREMENTS (Continued)

A7

Addressed in Section 5.0 Markup

3. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81, except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78 and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

<SR NOTES>

f. At least once per 18 months#****, during shutdown, by: A4

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service. LA3

SR 3.8.1.8 2. Verifying the diesel generator capability to reject a load of greater than or equal to 917.5 kw for diesel generator 1A, greater than or equal to 509.2 kw for diesel generator 1B, and greater than or equal to 1995 kw for diesel generator 1C while maintaining engine speed less than nominal plus 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less. LA4 A8 3.8 #68

SR 3.8.1.9 3. Verifying the diesel generator capability to reject a load of 3030-3130 kw*** for diesel generators 1A and 1B and 2500-2600 kw*** for diesel generator 1C without tripping. The generator voltage shall not exceed 4784 volts for diesel generator 1A and 1B and 5400 volts for diesel generator 1C during and following the load rejection.

M9

INSERT pf limit

SR 3.8.1.10 4. Simulating a loss of offsite power by itself, and:

L15

a) For divisions I and II:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

LA8

#For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

SR NOTE 1 ***Momentary transients due to changing bus loads shall not invalidate the test.

****Except 4.8.1.1.2.f.1 to be performed every refueling outage, for diesel generators 1A and 1B only. LA3

LAR 93-14R1

2) Verifying the diesel generator starts** on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the sequencing logic, and operates for greater than or equal to 5 minutes while its generator is loaded with the loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

b) For division III:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts** on the auto-start signal, energizes the emergency bus with the permanently connected loads within 13 seconds, energizes the auto-connected loads through the sequence logic, and operates for greater than or equal to 5 minutes while its generator is loaded with the loads. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

SR 3.8.1.14 ~~e)~~ Operating** with the diesel generator loaded to 3000-3100 kW*** for diesel generators 1A and 1B and 2500-2600 kW*** for diesel generator 1C for at least 60 minutes or until operating temperatures have stabilized. Within 5 minutes after completing this test, perform Surveillance Requirement 4.8.1.1.2.~~f.4.a)2) and b)2)~~ a. ⇐ <a.4 equivalent>

L11

L13

SR 3.8.1.11 ~~f.~~ Verifying that, on an ECCS actuation test signal without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. For diesel generator 1A and 1B, the generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal. For diesel generator 1C, the generator voltage and frequency shall not exceed a maximum of 5400 volts and 66.75 Hz and shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 13 seconds. The steady-state generator voltage and frequency shall be maintained within these limits during this test.

3.8 #73 ~~INSERT Item e & f~~ ~~(???)~~ e

<SR NOTES>

**All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading (>150 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

***Momentary transients due to changing bus loads shall not invalidate the test.

<SR 3.8.1.14 NOTE>

LAR 93-14R1

SR 3.8.1.15

~~20~~. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source, while the generator is loaded with its emergency loads, upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

SR 3.8.1.16

~~21~~. Verifying that, with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

SR 3.8.1.17

~~22~~. Verifying ~~that the automatic load sequence timers are OPERABLE with the interval between each load block~~ within ± 10% of its design ~~interval for diesel generators 1A, 1B and 1C~~. for each load sequence timer

L16

3.8 #15

~~23~~. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

A6

a) For Diesel Generators 1A and 1B:
1) Loss of control power to diesel control panel.
2) Starting air pressure below 150 psi.
3) Stop-solenoid energized.
4) Diesel in the maintenance mode (includes barring device engaged).
5) Overspeed trip device actuated.
6) Generator backup protection lockout relay tripped.

LA6

b) For Diesel Generator 1C:
1) Diesel generator lockout relays not reset.
2) Diesel engine mode switch not in "AUTO" position.
3) Diesel generator output breaker closed before start of diesel.
4) Diesel generator output breaker in racked-out position.
5) †Diesel generator regulator mode switch not in "AUTO" position.
6) Insufficient starting air pressure.
7) Loss of dc power to diesel generator controls.

LA6

LA2

g. By verifying the Division III diesel generator ambient room temperature to be ≥ 40°F:

1. At least once per 24 hours with the last reported room temperature ≥ 50°F, or

†Item 5) does not electrically block diesel generator from emergency starting; however, it will affect the loading and operation of the diesel.

LA6

LAR 93-14R1

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3.8 #85

72

24 hours M1

3.8 #25

L8 b. At least once per 92 days, and within 7 days after a battery ~~discharge with battery terminal voltage below 110 volts or after a battery~~ overcharge with battery terminal voltage above 144 volts, by verifying that:

3.8 #25

L8

SR 3.8.6.2 ~~1.~~ The parameters in Table 4.8.2.1-1 meet the Category B limits,

SR 3.8.4.2 ~~2.~~ There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohms, and

representative LA2

SR 3.8.6.3 ~~3.~~ The average electrolyte temperature of at least one out of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

SR 3.8.4.3 ~~1.~~ The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

L1

SR 3.8.4.4 ~~2.~~ The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

SR 3.8.4.5 ~~3.~~ The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohms and

SR 3.8.4.6 ~~4.~~ The battery charger will supply at least 300 amperes for chargers 1A and 1B and 50 amperes for charger 1C at a minimum of 130.2 volts for at least 8 hours.

A3

SR 3.8.4.7 ~~d.~~ At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design duty cycle when the battery is subjected to a battery service test, or

required LA3

2. The battery capacity is adequate to supply a dummy load of the following profile in accordance with IEEE 450 while maintaining the battery terminal voltage greater than or equal to 105 volts.

   a) Division I
   $\geq$ 671 amperes for the first 60 seconds
   $\geq$ 270 amperes for the next 9 minutes
   $\geq$ 336 amperes for the next 60 seconds
   $\geq$ 270 amperes for the next 228 minutes
   $\geq$ 451 amperes for the last 60 seconds

LAR 93-14R1

b) Division II
≥ 502 amperes for the first 60 seconds
≥ 261 amperes for the next 9 minutes
≥ 327 amperes for the next 60 seconds
≥ 261 amperes for the next 228 minutes
≥ 327 amperes for the last 60 seconds

LA3

c) Division III
≥ 53.2 amperes for the first 60 seconds
≥ 15.4 amperes for the next 119 minutes

A3

SR 3.8.4.8 ~~e~~. At least once per 60 months by verifying during shutdown that the battery capacity is at least 80% of the manufacturer's rating when

SR 3.8.4.7 NOTE subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

M5 3.8 #75 A3

SR 3.8.4.8 ~~f~~. At least once per 18 months, during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% 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.

LA4

LAR 93-14R1

ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

(LAR 93-14R1)

DISCUSSION OF CHANGES
CTS: 3.8.1.1 - A.C. SOURCES - OPERATING

ADMINISTRATIVE
(continued)

This comment number not used for this station.

3.8 #68 A.5 ~~These two possible values for the overspeed trip point are fixed by the design of the DG unit. The appropriate value (i.e., the most limiting) is presented in the proposed Technical Specifications. This presentation eliminates the basis for the accepted value from the Technical Specifications, moving it to the Bases. Since there is no difference in the requirement, this is an editorial presentation preference only.~~

A.6 The intent of this Surveillance ("...prevent diesel generator starting only when required...") is considered superfluous, in essence requiring the lockout to not be in effect unless the lockout condition or action of the operator is present. This is confirmed each time the DG experiences a demand for start - at least once per month. Removal of the specific 18 month Surveillance is therefore considered administrative.

A.7 The technical content of this requirement is being moved to another chapter of the proposed Technical Specifications in accordance with the format of the BWR Standard Technical Specification, NUREG-1434. Any technical changes to this requirement will be addressed with the content of the proposed chapter location.

A.8 This comment number not used for this station.

A.9 This comment number not used for this station.

A.10 This comment number not used for this station.

A.11 This comment number not used for this station.

A.12 The format of the proposed Technical Specifications does not include providing "cross references." Proposed LCOs adequately prescribe the Required Actions without such references. Therefore the existing reference to the various Specifications serves no functional purpose, and its removal is purely an administrative difference in presentation.

A.13 This comment number not used for this station.

A.14 Two Notes are provided to more clearly present the current methodology for conducting this surveillance. Since this is only a change in presentation of current practice, this change is considered administrative.

A.15 As discussed in USAR Section 8.3.1.1.3.6.1.2 and SER 8.4.8, the RBS load sequence design utilizes individual timers and permissive circuitry for each load which is automatically sequenced onto the standby buses. USAR Tables 8.3-2A, -2B and -

LAR 93-14RT

DISCUSSION OF CHANGES
CTS: 3.8.1.1 - A.C. SOURCES - OPERATING

TECHNICAL CHANGE - MORE RESTRICTIVE
(continued)

loss of all offsite AC power. Certain combinations of HPCS inoperability along with other ECCS systems (ADS and/or LPCS) may be in this category. Therefore, the proposed Technical Specifications require this verification even with the HPCS DG inoperable. This additional requirement added operational conservatisms not in the existing Technical Specifications.

M.4 Existing Technical Specifications allow both DG 1C, and DG 1A or 1B to be inoperable concurrently for 72 hours. The proposed ACTIONS for this Condition would limit the allowed outage time to 24 hours. After this 24 hours HPCS could be declared inoperable (this would allow the remainder of the 72 hours to be applied for DG 1A or 1B). This places a more restrictive time prior to having to declare the HPCS inoperable than the currently allowed 72 hours, in the event two DGs are inoperable.

M.5 This comment number not used for this station.

M.6 This comment number not used for this station.

M.7 This comment number not used for this station.

M.8 This comment number not used for this station.

M.9 Limitations on the operating power factor are added to load rejection tests and to the 24-hour run Surveillance. These limitations ensure the DG is conservatively tested at as close to accident conditions as reasonable.

3.8 #73

M.10 ~~Additional acceptance criteria are included in the proposed Technical Specifications. These functions are necessary for the system to perform its intended function.~~ e
This comment number not used for the station.

M.11 ~~The allowance to perform this test when "operating temperatures have stabilized" is removed. The test will only be able to be performed following a 1 hour run, even if temperatures have sooner stabilized. This eliminates any possible interpretations on which temperatures and what is "stabilized."~~ c

M.12 As with all other DG start requirements, this test is proposed to have added the acceptance criteria for voltage limits (upper and lower) and speed/frequency upper limit (lower limit included in the existing Surveillance). These acceptance criteria are consistent with all other DG start acceptance criteria.

M.13 In order for the consecutive test count to be valid for resetting the test frequency to 31 days, each test is to be

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.8.1.1 - A.C. SOURCES - OPERATING

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

If the original DG is restored to OPERABLE in less than 24 hours it would seem acceptable to allow a significantly longer time to complete the accelerated verification of the remaining DG. (If not restored to OPERABLE within 24 hours, then the accelerated verification would be required and is not suggested to be affected by this proposed change.) Once agreed that it is acceptably prudent to allow a longer time to complete the common-cause verification, then it should be acceptable to allow the plant program/procedures for Appendix B evaluations to complete the evaluation, and not require a TS directed action necessitating an exception to LCO 3.0.2, and potentially resulting in an unnecessary plant shutdown.

L.15 The phrase "actual or simulated" in reference to the test initiation signals, has been used in lieu of the surveillance requirement for verifying that each subsystem actuates on a "simulated" or "test" signal. This clarifies that satisfactory automatic system initiations for other than surveillance purposes can be used to fulfill the surveillance requirements. OPERABILITY is adequately demonstrated in either case since the subsystem itself can not discriminate between "actual" or "simulated."

3.8 #15

L.16 The diesel generator sequence loading calculation methodology adds load in load blocks for simplicity (as a typical example all the loads added between 20 & 30 seconds were summated at 20 seconds). The methodology was reexamined to evaluate the diesel loading anywhere within the time band of the timers from the lower limiting safety system setpoint to the upper limiting safety system setpoint. The specific conditions observed to be acceptable: 1) all timers could operate at their minimum lower limiting safety system setpoint to add load to the EDG at the earliest possible time frame and 2) all potentially overlapping loads could be combined to add the maximum load at one fixed time without overloading the diesel generators.

Graphs for each generator load sequencing have been composed with consideration to both upper and lower limits of each timer band to evaluate all potential loading conditions. The analogy of all potentially overlapping loads being combined to add the maximum load at one fixed time can easily be seen to be much less severe than the starting transient for the LPCS motor. The starting current for all load blocks have been reviewed, in no case does the current approach the starting current demonstrated in the shop margin test. The load connected to the diesel generators have been tabulated by a step load block method. For simplicity and conservatism in the calculation all loads are added at the beginning of the load block. Analysis for each load of the EDG being added at both the lower limiting

DISCUSSION OF CHANGES
CTS: 3.8.1.1 - A.C. SOURCES - OPERATING

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

safety system setpoint is well within the vendor tested load profile.

3.8 #15

As discussed in USAR Section 8.3.1.1.3.6.1.2 and SER 8.4.8, the RBS load sequence design utilizes individual timers and permissive circuitry for each load which is automatically sequenced onto the standby buses. USAR Tables 8.3-2A,-2B and -3, identify the specific loads associated with each standby bus and includes nomimal starting times for each load. The "LOAD BLOCKS" included in these USAR Tables were developed in Engineering Calculation G13.3E-192. This calculation grouped loads together in blocks of time and assumed for analytical purposes that each group represented a step change for the standby diesel generator. In reality, each load within these assumed load blocks has its own individual loading time.

The load blocks and associated time intervals in Calculation E-192 and USAR Table 8.3-2a, -2b and -3, cannot be directly correlated with the "LOAD BLOCKS" and "DESIGN INTERVALS" discussed in the subject surveillance requirement (SR).

However, although the "design" intervals between each "load block" referred to in the SR are not specifically identified in the design calculation, they are intrinsically defined by all the load sequence timer setpoints. Based on this fact, it is evident that if each sequenced load occurs at the proper time as analyzed in Calculation E-192, the "intervals between load blocks" refered to in the SR will be inherently satisfied. The proposed wording of the applicable ITS SR is necessary to reflect the load sequence logic design at RBS.

LAR 93-14(1)

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.2 When HPCS is required OPERABLE an additional flexibility is proposed associated with the requirement for its required source of power. The existing Technical Specification require only the HPCS DG be OPERABLE. The proposed Technical Specification provides an option for either the HPCS DG or a second offsite circuit (different than the circuit required for the remainder of the required components). This option for a second circuit in lieu of the HPCS DG is judged to provide similar or greater power reliability for the HPCS. Offsite circuit reliability has proven reliability at least as great as the typical DG unit.

L.3 Many of the currently required Surveillances involve tests that would require the DG to be paralleled to offsite power. This condition (the only required DG and the only required offsite circuit connected) presents a risk of a single fault resulting in a station blackout. The NRC has previously recognized this and provided surveillance exceptions to avoid this condition, but the direction has not been consistently applied. In an effort to consistently address this concern and to avoid potential conflicting Technical Specifications, the Surveillances which would require the DG to be connected to the offsite source are excepted from performance requirements. The exception does not take exception to the requirement for the DG to be capable of performing the particular function - just to the requirement to demonstrate it while that source of power is being relied on to support meeting the LCO.

3.8 #23

In addition, since the required DG is not required to be tested in parallel operation, this change also eliminates the requirement for the DG to be capable of returning to standby condition in the event an emergency signal is received while the DG is being tested. Since this is not a scenario for which the DG would be expected to respond, this capability is no longer required by Technical Specifications.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.8.2.1 - D.C. SOURCES - OPERATING

3.8 #85

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 In the event a battery experiences ~~a discharge (not a momentary transient as in the starting of a large load) which brings the battery terminal voltage below 110 volts,~~ an overcharge e confirmation of continued battery OPERABILITY should be made sooner than 7 days. The proposed change to ~~24~~ 72 hours allows sufficient time to plan for an unscheduled surveillance and complete the performance of the surveillance of cell parameters without undo haste. Since this confirmation of OPERABILITY is revised to make that determination sooner, the change is conservative.

M.2 The allowance to correct the Category B limit for temperature is being proposed for deletion based on the BWR Standard Technical Specification, NUREG-1434, presentation and IEEE-450 recommendations.

M.3 Limitations are proposed to be imposed on this allowance: the utilization of charging current limited to 7 days, and a requirement to measure actual specific gravities at the end of this period. These restrictions will assure excessive reliance on charging current is not made.

M.4 Proposed Required Action A.1 requires a more immediate check that pilot cell electrolyte level and float voltage are within limits. Required Action A.2 proposes a periodic re-verification that all cell parameters are within limits. These restrictions provide added assurance of adequate battery capabilities for the period allowed to completely restore the cell parameters.

3.8 #75

M.5 ~~The frequency for Surveillance 4.8.2.1.f has been changed from 18 months to 12 months. Since the change will require the test on a more frequent bases, the change is considered more restrictive.~~ e This comment number not used for this station.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 The details relating to system design and purpose have been relocated to the Bases. The design features and system operation are also described in the USAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program in Chapter 5 of the TS.

LA.2 IEEE-450 (from which these tests are derived) requires only "representative" cells be measured for temperature; every sixth cell is provided as a suggestion only. Therefore, the detail relating to the plant specific determination of "representative" is proposed to be relocated to the Bases.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 3.8.2.1 - D.C. SOURCES - OPERATING

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

L.6 This comment number not used for this station.

L.7 Based on discussions with the IEEE-450 Battery Working Group, the NRC has found a revision to the restoration time, from 7 days to 31 days, acceptable (as documented in the BWR Standard Technical Specification, NUREG-1434). In conjunction with the periodic monitoring required by the proposed Required Action A.2, this change does not represent a significant relaxation in battery capability.

3.8 #25

L.8 Battery terminal corrosion is not significantly affected by a battery discharge or overcharge. Therefore, the proposed Frequency does not include the performance of the SR following these events.

LAR 93-14R1

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.8.1.1 - A.C. SOURCES - OPERATING

"L16" CHANGE

3.8 #15

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Sequence timers are necessary to support the equipment used to mitigate the consequences of an accident, but the sequence timers are not considered as the initiator of any previously analyzed accident. As such the proposed revision to the Surveillance Requirements will not increase the probability of any accident previously evaluated. The proposed Surveillance Requirements continue to provide adequate assurance of OPERABLE sequence timers and therefore, does not involve any increase to the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the sequence timers continues to be determined in the same manner.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.8.1.2 - A.C. SOURCES - SHUTDOWN

"L3" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   A power supply is necessary to support the equipment used to mitigate the consequences of an accident, but the power supply is not considered as the initiator of any previously analyzed accident. As such the proposed revision to the Surveillance Requirements will not increase the probability of any accident previously evaluated. The proposed SRs continue to provide adequate assurance of OPERABLE DGs and available offsite circuits and therefore, does not involve any increase to the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change removes requirements for paralleling the required DG to the required offsite circuit and demonstration of capabilities that the DG is not expected to be needed to respond to. Omitting these required tests represents a significant improvement in the margin of safety by removing the potential for a single fault to affect both required power sources.

3.8 #23

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 3.8.2.1 - D.C. SOURCES - OPERATING

3.8 #25

"L8" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The DC electrical power subsystems are used to support mitigation of the consequences of an accident, but they are not considered as the initiator of any previously analyzed accident. As such the removal of this surveillance following a battery discharge or overcharge will not increase the probability of any accident previously evaluated. The proposed SR Frequency continues to provide adequate assurance of OPERABLE batteries since the batteries connections are not significantly affected by a discharge or overcharge. Therefore, the proposed change does not involve any increase to the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the SR Frequency will continue to assure the batteries are not degraded beyond current accepted allowances.

10/17/94

LAR 93-14 R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.8 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14 R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.~~9~~ 8	-----NOTES----- 1. This Surveillance shall not be performed in MODE 1 or 2. However, ~~2.~~ credit may be taken for unplanned events that satisfy this SR. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ [0.9]. ---------- Verify each DG ~~operating at a power factor ≤ [0.9]~~ rejects a load ~~≥ [1200] kW for [Division 1 and ≥ [550] kW for Division 2] DGs and ≥ [2180] kW for [Division 3] DG,~~ ~~and~~ equivalent to the single largest post accident load greater than or equal to its associated ~~a.~~ following load rejection, the ~~frequency is ≤ [69] Hz;~~ 66.75 engine speed is maintained less than nominal plus 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is lower. ~~b. Within [3] seconds following load rejection, the voltage is ≥ [3744] V and ≤ [4576] V; and~~ ~~c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.~~ C3 P8 3.8 #68 B1 3.8 #47 C42 INSERT 9A	C4 ~~[18 months]~~ B1 P27
SR 3.8.1.~~10~~ 9	-----NOTES----- ~~1.~~ This Surveillance shall not be performed in MODE 1 or 2. However, ~~2.~~ credit may be taken for unplanned events that satisfy this SR. ---------- C4 Verify each DG operating at a power factor ≤ [0.9] does not trip and voltage is maintained ≤ ~~[6000]~~ 4784 for DG 1A and DG 1B and ≤ 5400 V for DG 1C V during and following a load rejection of a load ≥ ~~[4600] V~~ 3030 kW and ≤ ~~[5740]~~ 3130 kW for ~~[Division 1 and 2] DGs~~ DG 1A and 1B and ≥ ~~[3300]~~ 2500 kW and ≤ ~~[3500]~~ 2600 kW for ~~[Division 3] DG~~ DG 1C. B1	~~[18 months]~~ C19 B1

(continued)

LAR 95-?? 1

INSERT 9A

--------------------NOTES--------------------
1. Momentary transients outside the load and power factor ranges do not invalidate this test.

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3.8
#48

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.~~12~~ 11	--------------------NOTES-------------------- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. However, (C4) ~~3.~~ ~~C~~credit may be taken for unplanned events that satisfy this SR. --- Verify on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal each DG auto-starts from standby condition and: (P 28) a. For DG 1C during the auto-start maintains voltage ≤ 5400 V and frequency ≤ 66.75 Hz; (B1) ~~a.~~ b In ≤ [10] seconds after auto-start and during tests, achieves voltage ≥ ~~[3744]~~ 3740 V and ≤ ~~[4576]~~ 4580 V; ~~b.~~ c In ≤ [10] seconds after auto-start and during tests, achieves frequency ≥ [58.8] Hz and ≤ [61.2] Hz; and for DG1A and DG1B and ≤ 13 seconds for DG 1C (B1) ~~c.~~ d Operates for ≥ [5] minutes; ~~and~~ . (P38) ~~d.~~ e Permanently connected loads remain energized from the offsite power system; and ~~e.~~ f Emergency loads are ~~energized for~~ auto-connected ~~through the automatic load sequencer~~ to the offsite power system. (C34)	(B1) [18 months] 3.8 #73

(continued)

3.8 #73

LAR 93-14R1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.~~14~~ 13	-------------------NOTES------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. ~~However,~~ ~~3.~~ Credit may be taken for unplanned events that satisfy this SR. --- Verify each DG ~~DG 1A and 1B~~ operating at a power factor ≤ ~~[~~0.9~~]~~ ~~for Division 1 and 2 DGs, and ≤ [0.9] for Division 3 DG~~, operates for ≥ 24 hours: ~~[ ]~~ a. 1. For ≥ ~~[~~2~~]~~ hours loaded ≥ ~~[ ]~~ 2850 2750 kW and ≤ ~~[ ]~~ kW; ~~for Division 1 and 2 DGs, ≥ [3630] kW and ≤ [3830] kW for Division 3 DG;~~ and b. 2. For the remaining hours of the test loaded ~~≥ [3744] kW and ≤ [4576] kW for Division 1 and 2 DGs, and~~ ≥ ~~[3300]~~ 2500 kW and ≤ ~~[3600]~~ 2600 kW~~ for Division 3 DG~~.	C4 3.8 #51 P11 INDENT	[18 months] B1

(continued)

a. ~~For~~ DG 1A and DG 1B loaded ≥ 3030 kW and ≤ 3130 kW; and P2

b. For DG 1C :

LAR # 1

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources—Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

C6 3.8 #65

a. One qualified circuit Connected between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems—Shutdown"; and

b. One diesel generator (DG) capable of supplying one division of the Division ~~1~~ I or ~~2~~ II onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10; and

3.8 #66 C6

c. One qualified circuit between the offsite transmission and the Division ~~3~~ III onsite Class 1E electrical power distribution subsystem other than the circuit in LCO 3.8.2.a, or the Division ~~3~~ III DG ~~when~~ capable of supplying the Division ~~3~~ III onsite Class 1E AC electrical power distribution subsystem, when the Division III onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.10.

APPLICABILITY: MODES 4 and 5,
B1 During movement of irradiated fuel assemblies in the primary ~~or secondary~~ containment or Fuel Building.

------------ NOTE ------------
LCO 3.0.3 is not applicable.
------------------------------ 3.8 #67

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C6 A. LCO item a not met. ~~One or more required offsite circuits inoperable.~~	------------NOTE------------ Enter applicable Condition and Required Actions of LCO 3.8.10, ~~with one~~ when any required division is de-energized as a result of Condition A. ----------------------------	C32
	A.1 Declare affected required feature(s) with no offsite power from a required circuit available, inoperable. C6	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
B1	A.2.2 Suspend movement of irradiated fuel assemblies in the primary ~~and secondary]~~ containment and Fuel Building.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
	A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.2.1	-------------------NOTE------------------- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.8 through SR 3.8.1.~~11~~ 10, SR 3.8.1.~~13~~ 12 through SR 3.8.1.~~16~~ 15, ~~[~~SR 3.8.1.~~18~~ 17~~]~~, and SR 3.8.1.~~19~~ 18. C7 P33 --- For AC sources required to be OPERABLE, the following ~~SRs of Specification 3.8.1, except SR 3.8.1.17 and SR 3.8.1.20, are applicable~~ SR 3.8.1.7, 16, 19 SR 3.8.1.1 SR 3.8.1.2 SR 3.8.1.3 SR 3.8.1.4 SR 3.8.1.5 SR 3.8.1.6 SR 3.8.1.8 SR 3.8.1.9 SR 3.8.1.10 SR 3.8.1.11 SR 3.8.1.12 SR 3.8.1.13 SR 3.8.1.14 SR 3.8.1.15 SR 3.8.1.17 SR 3.8.1.18 3.8 #82	~~In accordance with applicable SRs~~

LAR 93-14R1

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel oil level ~~:~~ ~~1. For [DG 11 or 12],~~ < 45,495 ~~[62,000]~~ gal and ≥ 38,996 ~~[49,000]~~ gal. ~~2. For [DG 13], < [41,200] gal and ≥ [38,500] gal.~~	A.1 Restore fuel oil level to within limits. B1	48 hours
B. One or more DGs with lube oil inventory ~~within limits and:~~ C8 3.8 #6 B1 1. For [DG 1A or 1B], < 367 [ ] gal and ≥ 350 ~~[425]~~ gal; and 2. For [DG 1C], < 295 [ ] gal and ≥ 265 [ ] gal.	B.1 Restore lube oil inventory to within limits.	48 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1 (B1)	Verify each fuel oil storage tank contains ≥ 45,495 ~~[62,000]~~ gal of fuel. ~~for [DGs 11 and 12]; and~~ ~~b. ≥ [41,200] gal of fuel for [DG 13].~~	31 days
SR 3.8.3.2 (B1)	Verify lube oil inventory is: a. ≥ 367 ~~[ ]~~ gal for ~~[~~DGs 1A ~~11~~ and 1B ~~12~~~~]~~ and b. ≥ 295 ~~[ ]~~ gal for ~~[~~DG 1C ~~13~~~~]~~.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each required (P29) DG air start receiver pressure is ≥ 160 ~~[225]~~ psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	~~[~~31~~]~~ days (B1)
SR 3.8.3.6	For each fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years

3.8 #63

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.4.8 C4	------------------NOTES------------------ ~~1.~~ This Surveillance shall not be performed in MODE 1, 2, or 3. However, ~~2.~~ ~~C~~credit may be taken for unplanned events that satisfy this SR. --	
B1	Verify battery capacity is ≥ ~~[80]~~% of the manufacturer's rating when subjected to a performance discharge test. 3.8 #75	60 months AND -----NOTE------ Only applicable when battery shows degradation or has reached ~~[85]~~% of expected life -------------- ~~12~~ 18 months

LAR 93-0001

INSERT 29A

3.8
#75

12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating

AND

24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

7/8/94

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources—Shutdown

INSERT 30A

3.8 #98

LCO 3.8.5 ~~DC electrical power subsystem(s) shall be OPERABLE to support the electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems—Shutdown."~~

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the [primary ~~or secondary]~~ containment or Fuel Building.

B1

3.8 #67

-------NOTE-------
LCO 3.0.3 is not applicable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies in the [primary ~~or secondary]~~ containment and Fuel Building.	Immediately
B1	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
		(continued)

LAR 93-14R1

INSERT 30A

3.8 #98

The following shall be OPERABLE:

a. One Class 1E DC electrical power subsystem capable of supplying one division of the Division I or II onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown";

b. One Class 1E battery or battery charger, other than the DC electrical power subsystem in LCO 3.8.5.a, capable of supplying the remaining Division I or II onsite Class 1E DC electrical power distribution subsystem(s) when required by LCO 3.8.10; and

c. The Division III DC electrical power subsystem capable of supplying the Division III onsite Class 1E DC electrical power distribution subsystem, when the Division III onsite Class 1E DC electrical power distribution subsystem is required by LCO 3.8.10.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	--------------------NOTE-------------------- The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. -- 3.8 #84 For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6	In accordance with applicable SRs

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND ~~Once within 24 hours after battery discharge < [110] V~~ B1 ~~AND~~ 3.8 #85 Once within ~~24~~ 72 hours after battery overcharge > ~~[150]~~ 144 V B1
SR 3.8.6.3 B1	Verify average electrolyte temperature of representative cells is ≥ [60]°F.	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters—Shutdown

INSERT 38A

3.8 #98

LCO 3.8.8 ~~Inverter(s) shall be OPERABLE to support the onsite Class 1E AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems—Shutdown."~~

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment or Fuel Building.

3.8 #67 B1

NOTE
LCO 3.0.3 not applicable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more [required] inverters inoperable. (B1)	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
(B1)	A.2.2 Suspend handling of irradiated fuel assemblies in the [primary ~~or secondary~~] containment and Fuel Building.	Immediately
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
		(continued)

LAR 93-14X1

INSERT 38A

3.8 #98

One Divisional inverter shall be OPERABLE capable of supplying one division of the Division I or II onsite Class 1E uninterruptible AC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown".

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems—Operating

LCO 3.8.9 Division I and II [Division ~~1~~ I], [Division ~~2~~ II], and [Division ~~3~~ III] AC and DC, [and AC vital bus] electrical power distribution subsystems shall be OPERABLE. B1

APPLICABILITY: MODES 1, 2, and 3.

INSERT 40 A C14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C22 B1 A. ~~One~~ One or more [Division ~~1 or 2~~ I or II] AC electrical power distribution subsystems inoperable. 3.8 #72	A.1 B1 Restore [Division ~~1~~ I and ~~2~~ II] AC electrical power distribution subsystems to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
B1 B. One or more [Division ~~1~~ I or ~~2~~ II] AC vital bus distribution subsystems inoperable.	B.1 Restore [Division ~~1~~ I and ~~2~~ II] AC vital bus distribution subsystems to OPERABLE status.	~~2~~ 8 hours B1 AND 16 hours from discovery of failure to meet LCO

(continued)

3.8 #72

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more [Division ~~1 or 2~~ I or II] DC electrical power distribution subsystems inoperable. (B1)	C.1	Restore [Division I and II] DC electrical power distribution subsystems to OPERABLE status. (B1)	2 hours AND 16 hours from discovery of failure to meet LCO
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 AND D.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours
E. One or more [Division ~~2~~ III] AC, DC, or AC vital bus electrical power distribution subsystems inoperable. (B1)	E.1	Declare High Pressure Core Spray System [and ~~2~~ Standby Service Water System] pump 2C inoperable.	Immediately (B1)

INSERT 41A (C13)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1 (B1)	Verify correct breaker alignments and voltage to [required] AC, DC, [and AC vital bus] electrical power distribution subsystems.	7 days

INSERT 41A

CONDITION	REQUIRED ACTION	COMPLETION TIME
3.8 #31 ~~E~~. F. Two or more divisions with inoperable distribution subsystems that result in a loss of function.	~~E~~ F.1 Enter LCO 3.0.3.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems—Shutdown

Division I and II

III

LCO 3.8.10 The necessary portions of the Division ~~1~~ I, Division ~~2~~ II, and Division ~~3~~ AC, DC, [and AC vital bus] electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE. P11

B1

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment. or Fuel Building

3.8 #67

B1

NOTE
LCO 3.0.3 is not applicable

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, ~~or~~ [or AC vital bus] electrical power distribution subsystems inoperable. (B1)	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND (C28)	
	A.2.2 Suspend ~~handling~~ movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment. and Fuel Building (B1)	Immediately
	AND	
		(continued)

BASES

LCO (continued)

P13

3.8 #71

INSERT B4A

C44

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds for DG 1A and DG 1B and 13 seconds for DG 1C. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot, and DG in standby with engine at ambient conditions, ~~and DG operating in parallel test mode~~.

P1

Proper sequencing of loads, including tripping of nonessential loads, is a required function for both offsite circuit and DG OPERABILITY on Divisions I and II.

The AC sources in one division must be separate and independent (to the extent possible) of the AC sources in the other division(s). For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical.

APPLICABILITY

P1

The AC sources [redacted] are required to be OPERABLE in MODES 1, 2, and 3 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

P11

A Note has been added taking exception to the Applicability requirements for Division ~~3~~ III sources, provided the HPCS System and SSW pump 2C is declared inoperable. This exception is intended to allow declaring of the ~~Division 3~~ HPCS System inoperable either in lieu of declaring the Division ~~3~~ III source inoperable, or at any time subsequent to entering ACTIONS for an inoperable Division ~~3~~ III source. This exception is acceptable since, with the ~~Division 3~~ HPCS System inoperable and the associated ACTIONS entered, the Division ~~3~~ III AC sources provide no additional assurance of meeting the above criteria.

(continued)

LAR 93-14R1

INSERT B4A

3.8 #71

Additional DG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~8~~ 7 (continued)

B1

safety systems. ~~Note 2 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

3.8 #76 --- INSERT B21A

SR 3.8.1.~~9~~ 8 P33

3.8 #68

P27 C28 Core

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load ~~without exceeding predetermined voltage and frequency and~~ while maintaining a specified margin to the overspeed trip. The load referenced for DG ~~11~~ 1A is the ~~1200~~ 922.5 kW low pressure ~~coolant~~ spray pump; for DG ~~12~~ 1B, the ~~550~~ 462.2 kW residual heat removal (RHR) pump; and for DG ~~13~~ 1C the ~~2100~~ 1995 kW HPCS pump. The Standby Service Water (SSW) pump values are not used as the largest load since the SSW supplies cooling to the associated DG. If this load were to trip, it would result in the loss of the DG. As required by IEEE-308 (Ref. 13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. For the ~~Grand Gulf Nuclear~~ River Bend Station the~~se~~ values ~~are the same~~ results from the first criteria lower.

P8 P11 P11 P23

~~The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of the 5 second load sequence interval associated with sequencing of this largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection.~~ The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

P27

C3

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as

(continued)

LAR 93-14R1

INSERT B21A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

LAR 93-14R1

3.8 #69 — offsite power

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~13~~ 8 (continued)

Note 2 requires that, if synchronized to ~~the bus~~ offsite power

possible, testing ~~should~~ be performed using a power factor ≤ ~~[~~0.9~~]~~. This power factor is chosen to be representative of the actual design basis inductive loading that the DG could experience.

B1

C3

This SR has been modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. ~~Note 2 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

3.8 #76

C4

INSERT B22A

B1

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable;~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~

~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.~~15~~ 9 P33

This Surveillance demonstrates the DG capability to reject a full load, i.e., maximum expected accident load, without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide DG damage protection. While the

C39

(continued)

LAR 93-1421

INSERT B22A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~10~~ 9 (continued)

DG is not expected to experience this transient during an event, and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

3.8 #48

B1

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ ~~[~~0.9~~]~~. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

B1

C4

The ~~[~~18 month~~]~~ Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

~~Note 1 states that momentary transients due to changing bus loads do not invalidate this test~~

C4

This SR has been modified by ~~two~~ a Notes. The reason for the Note ~~2~~ is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. [redacted] Credit may be taken for unplanned events that satisfy this SR.

INSERT B23A

3.8 #76

B1

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable;~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~

~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

(continued)

LAR 93-14R1

INSERT B23A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.~~11~~ 10 P33

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the Division I and II nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

C33

P13

The DG auto-start time of 10 seconds for DG 1A and DG 1B and 13 seconds for DG 1C is derived from requirements of the accident analysis to respond to a design basis large break LOCA. ~~The frequency should be restored to within 2% of nominal following a load sequence step.~~ The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

C40

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, ~~high pressure injection~~ systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

3.8 #87

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

B1

(continued)

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~11~~ 10 (continued)

C4

This SR is modified by ~~three~~ two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for ~~[Division 1 and 2 DGs]~~ DG1A and DG1B. [For the ~~[Division 3]~~ DG 1C, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by ~~the lubricating oil~~ an immersion heater and circulates through the system by natural circulation.] The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems.

B1

C37

COPY/REPEAT AS INSERT B25A to Bases pgs 17, 26, 32, 33

C4

~~Note 3 acknowledges that~~ credit may be taken for unplanned events that satisfy this SR.

3.8 #76

INSERT B25A

SR 3.8.1.~~12~~ 11 P33

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (~~[10]~~ seconds for DG1A and DG1B and 13 seconds for DG1C) from the design basis actuation signal (LOCA signal) and operates for ≥ ~~[5]~~ minutes. The ~~[5]~~ minute period provides sufficient time to demonstrate stability. ~~SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ECCS signal without loss of offsite power.~~

P28 P33 1.f 1.e P33 P28

B1

3.8 #73

~~The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the loading logic for loading onto offsite power. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In~~

(continued)

INSERT B25A

3.8 #716

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~22~~ 11 (continued)

3.8 #73

~~lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.~~

offsite power C18

The Frequency of ~~[~~18 months~~]~~ takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the ~~[~~18 month~~]~~ Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

B1

C4

This SR is modified by ~~three~~ two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. ~~Note 3 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

FROM PAGE B 3.8-25 C37

INSERT B25A

C2

INSERT B26A

3.8 #76

SR 3.8.1.~~23~~ 12 P33

This Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature) are bypassed on ~~a loss of voltage signal concurrent with~~ an ECCS initiation test signal and critical protective functions ~~(engine overspeed, generator differential current, and low lube oil pressure)~~ trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide ~~an~~ alarms on an abnormal engine condition. ~~This~~ These alarms provides the operator with ~~sufficient time~~ necessary information to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

P13

C45

C28

(continued)

LAR 93-14R1

INSERT B26A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~13~~ 12 (continued)

B1

The ~~[18 month]~~ 24 Frequency is based on engineering judgment, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the ~~[18 month]~~ 24 Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

B1

3.8 #76

C4

The SR is modified by ~~two~~ a Notes. The reason for ~~Note 2~~ the Note is that performing the Surveillance removes a required DG from service. ~~Note 2 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

INSERT B 27B

B1

~~[Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable;~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~

~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.]~~

SR 3.8.1.~~14~~ 13 P33

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours—22 hours of which is at a load equivalent to the continuous rating of the DG, and 2 hours of which is at a load equivalent to 110% of the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelube and warmup, discussed in SR 3.8.1.2,

P2

INSERT B27A

(continued)

LAR 93-14R1

INSERT B27A

An exception to the loading requirements is made for DG 1A and DG 1B. DG 1A and DG 1B are operated for 24 hours at a load greater than or equal to the maximum expected post accident load. Load carrying capability testing of the Transamerica Delaval Inc. (TDI) diesel generators (DG 1A and DG 1B) has been limited to a load less than that which corresponds to 201 psig brake mean effective pressure (BMEP). Therefore, full load testing is performed at a load ≥ 3030 kW but < 3130 kW.

INSERT B27B

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~15~~ 13 (continued)

and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

B1

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor ≤ ~~[~~0.9~~]~~. This power factor is chosen to be representative of the actual design basis inductive loading that the DG could experience.

B1

The ~~[~~18 month~~]~~ Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

C4

This Surveillance is modified by ~~three~~ two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. Similarly, momentary power factor transients above the limit do not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that would challenge continued steady state operation and, as a result, plant safety systems. ~~Note 3 acknowledges that~~ credit may be taken for unplanned events that satisfy this SR.

C4

3.8 #76

INSERT B28A

P33

SR 3.8.1.~~16~~ 14

for DG 1A and DG1B and within 13 seconds for DG 1C

P13

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The ~~[illegible]~~ time ~~is~~ requirements a derived from the requirements of the accident analysis to respond to a design basis large break LOCA.

B1

The ~~[~~18 month~~]~~ Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

(continued)

LAR 93-14R1

INSERT B28A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~15~~ 14 (continued)

This SR has been modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least ~~2 hours~~ 1 hour at full load conditions prior to performance of this Surveillance, and longer if necessary to stabilize the operating temperature, is based on manufacturer recommendations for achieving hot conditions. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

P13

SR 3.8.1.~~16~~ 15 P33

3.8 #74 P23 C28

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and ~~automatic~~ load transfer from the respective DG to ~~the~~ each required offsite ~~source~~ power source can be made and that the DG can be returned to ready-to-load status when offsite power is restored. It also ensures that the ~~auto-start~~ undervoltage logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready-to-load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto-close signal on bus undervoltage, and the load sequence timers are reset.

3.8 #74 P11

INSERT B29A

B1

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.~~b~~ a.(6), and takes into consideration plant conditions required to perform the Surveillance.

C38

This SR is modified by ~~two~~ a Notes. The reason for ~~Note 1~~ the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. ~~Note 2 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

C4

INSERT B29B

3.8 #76

(continued)

INSERT B29A

3.8 #74

Portions of the synchronization circuit are associated with the DG and portions with the respective offsite circuit. If a failure in the synchronization requirement of the Surveillance occurs, depending on the specific affected portion of the synchronization circuit, either the DG or the associated offsite circuit is declare inoperable.

INSERT B29B

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.~~17~~ 16 P33

Demonstration of the test mode override ensures that the DG availability under accident conditions is not compromised as the result of testing. Interlocks to the LOCA sensing circuits cause the DG to automatically reset to ready-to-load operation if an ECCS initiation signal is received during operation in the test mode. Ready-to-load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading is not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

B1 The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

C4 This SR has been modified by ~~two~~ a ~~Notes~~ Note. The reason for the Note ~~1~~ is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. ~~Note 2 acknowledges that~~ C credit may be taken for unplanned events that satisfy this SR.

INSERT B30A

3.8 #76

SR 3.8.1.~~18~~ 17 P33

C41 ~~As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), each DG 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 the~~

(continued)

LAR 93-14R1

INSERT B30A

3.8
#76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

P33

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~18~~ 17 (continued)

C41

~~DGs to their respective bus, all loads are shed except load~~ center feeders and those motor control centers that power Class 1E loads (referred to as "permanently connected" loads). Upon reaching 90% rated voltage and ~~frequency, the DGs are then connected to their respective bus.~~ Loads are ~~then~~ sequentially connected to the bus by the load sequencing ~~panel~~ logic. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the ~~DGs~~ bus power supply due to high motor starting currents. The ~~[10]%~~ load sequence time ~~interval~~ tolerance ensures that sufficient time exists for the ~~DG~~ to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. ~~Regulatory Guide 1.9 (Ref. 3)~~ Reference 2 provides a summary of the automatic loading of ESF buses.

P11 P15 B1 C38

B1

The Frequency of ~~[~~18 months~~]~~ is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2); takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

3.8 #76

This SR is modified by ~~two~~ a Note~~s~~. The reason for the Note ~~1~~ is that performing the Surveillance during these MODES would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. ~~Note 2 acknowledges that~~ Credit may be taken for unplanned events that satisfy this SR.

C4

INSERT C31A

B1

~~[Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

~~a. Performance of the SR will not render any safety system or component inoperable;~~

~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~

(continued)

LAR 93-14R1

INSERT B31A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SURVEILLANCE REQUIREMENTS

B1

~~SR 3.8.1.18 (continued)~~

~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

a

SR 3.8.1.~~19~~ 18 P33

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

B1

The Frequency of [18 months] takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of [18 months].

C4

This SR is modified by ~~three~~ two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. ~~Note 3 acknowledges that~~ credit may be taken for unplanned events that satisfy this SR.

<FROM PAGE 3.8-25> C37

INSERT B25A

C4

INSERT B32A --- 3.8 #76

SR 3.8.1.~~20~~ 19 P33

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this

(continued)

LAR 93-14R1

INSERT B32A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

LAK 93-1421

BASES

LCO (continued)

power source is available to provide electrical power support, assuming a loss of the offsite circuit. Similarly, when the high pressure core spray (HPCS) is required ~~to be~~ OPERABLE, a separate offsite circuit to the Division ~~3~~ III Class 1E onsite electrical power distribution subsystem, or an OPERABLE Division ~~3~~ III DG, ensure an additional source of power for the HPCS. INSERT B 37A Together, OPERABILITY of the required offsite circuit(s) and DG(s) ensure the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, reactor vessel draindown).

P11

C6

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective ESF bus(es), and accepting required loads during an accident. Qualified offsite circuits are those that are described in the ~~F~~USAR and are part of the licensing basis for the plant. ~~[~~The offsite circuit consists of incoming breaker and disconnect to the respective ~~service~~ preferred station transformers ~~(1 and 2)~~ 1C and 1D, the ~~(1 and 2)~~ service transformers, ~~ENS SSF transformers (1 and 2)~~ and the respective circuit path including feeder breakers to all 4.16 kV ESF buses required by LCO 3.8.10.~~]~~

P10

P15

The required DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within ~~[~~10~~]~~ seconds for DG1A and DG1B and 13 seconds for DG1C. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot, DG in standby with the engine at ambient conditions, ~~and DG operating in parallel test mode.~~ Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

P13

C44

Proper load sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. ~~[~~In addition, proper sequencer operation is an integral part of offsite circuit and DG OPERABILITY ~~if~~ since its inoperability ~~in any way~~ impacts ~~on~~ the ability to start and maintain energized any loads required OPERABLE by LCO 3.8.10.~~]~~

B1

3.8 #71

(continued)

LAR 93-14R1

BASES

P23

AC electrical power distribution subsystems

LCO (continued)

B1

It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required ~~divisions~~. ~~[No fast transfer capability is required for offsite circuits to be considered OPERABLE.]~~

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

APPLICABILITY

B1

The AC sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment or Fuel Building provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

INSERT B38A

3.8 #67

A.1

An offsite circuit is considered inoperable if it is not available to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.10, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential

(continued)

(LAR 93-14R1)

INSERT B38A

3.8 #67

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

Pursuant to LCO 3.0.6, the Distribution System ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to ~~one~~ any required ESF bus, ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a division is de-energized. LCO 3.8.10 provides the appropriate restrictions for the situation involving a de-energized division.

C6 C32

C.1

C6

When the HPCS is required to be OPERABLE, and the additional required Division ~~3~~ III ~~DG~~ AC source is inoperable, the required diversity of AC power sources to the HPCS is not available. Since these sources only affect the HPCS, the HPCS is declared inoperable and the Required Actions of the affected Emergency Core Cooling Systems LCO entered.

P11 C6

In the event all sources of power to Division ~~3~~ III are lost, Condition A will also be entered and direct that the ACTIONS of LCO 3.8.10 be taken. If only the Division ~~3 DG~~ III additional required AC source is inoperable, and power is still supplied to HPCS ~~by the circuit meeting the LCO item a requirement~~, 72 hours is allowed to restore the ~~DG~~ AC source to OPERABLE. This is reasonable considering HPCS will still perform its function, absent an additional single failure.

3.8 #66

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.1.9 is not required to be met since only one offsite circuit is required to be OPERABLE. C7

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, and 3. SR 3.8.1.~~17~~ 16 is not required to be met because the required OPERABLE DG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.20 19 is excepted because starting independence is not required with the DG(s) that is not required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

P33 P33

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1 (continued)

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during the performance of SRs, and preclude de-energizing a required 4.16 KV ESF Bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG is required to be OPERABLE.

REFERENCES

None.

3.8 #88

BASES

APPLICABILITY (continued)

starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

INSERT B 44A - - - 3.8 #89

A.1

In this Condition, the 7 day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. These circumstances may be caused by events such as:

a. Full load operation required after an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C28

B.1

With lube oil inventory less than required ~~< 500 gal~~, sufficient lube oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

(continued)

INSERT B44A

3.8
#89

The ACTIONS are modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.2 (continued)

the level reaching the manufacturer's recommended minimum level.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run times are closely monitored by the plant staff.

SR 3.8.3.3

C43

of fuel oil prior to addition to the storage tanks

The tests ~~listed below~~ are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate detrimental impact on diesel engine combustion and operation. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between ~~receipt~~ of new fuel and ~~conducting the tests~~ to exceed 31 days. The tests, limits, and applicable ASTM Standards ~~are as follows:~~

the sample (and corresponding results)

addition of new fuel oil to the storage tanks

for the tests listed in the Diesel Fuel Oil Testing Program of specification 5.5.9

5.0 #104

a. ~~Sample the new fuel oil in accordance with ASTM D4057-[81]~~ (Ref. 6); C16

b. Verify in accordance with the tests specified in ASTM D975-[81] (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 ~~and~~ ≤ 0.89 ~~for~~ ; an API gravity at 60°F of ≥ 27° and ≤ 39°; ~~a~~ kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes ~~, and a flash point of ≥ 125°F~~; and C19

P35 INSERT B47A

~~c.~~ e Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[82] (Ref. 6). B1

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO since the fuel oil is not added to the storage tanks.

C43 ~~Within 31 days~~ following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

properties specified in Table 1 of ASTM D975-[B1] (Ref. 6) are met for new fuel oil when tested in accordance with ASTM D975-[81] (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1522-[79] (Ref. 6) or ASTM D2622-[82] (Ref. 6). The 31 day period is acceptable because the fuel oil properties of interest, even if not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

B1

C43 INSERT B48A

Fuel oil degradation during long term storage shows up as an increase in particulate, mostly due to oxidation. The presence of particulate does not mean that the fuel oil will not burn properly in a diesel engine. However, the particulate can cause fouling of filters and fuel oil injection equipment, which can cause engine failure.

P35 INSERT B48B

Particulate concentrations should be determined in accordance with ASTM D2276-[78], Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. ~~[For those designs in which the total volume of stored fuel oil is contained in two or more interconnected tanks, each tank must be considered and tested separately.]~~ B1

B1

The Frequency of this Surveillance takes into consideration fuel oil degradation trends indicating that particulate concentration is unlikely to change between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. ~~The system design requirements provide for a minimum of five engine start cycles without recharging. [A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.]~~ The pressure specified in this SR ~~is intended to~~ reflects the ~~lowest~~ value at which ~~the [five] starts~~ this can be accomplished, but is not so low high as to result in failing the limit due to normal cycling of the recharge compressor.

P37 INSERT C48C

P23

3.8 #91

(continued)

INSERT B48A

5.0 #104

5.5.9

These additional analyses are required by Specification ~~5.7.15~~, Diesel Fuel Oil Testing Program, to be performed within 31 days following sampling and addition. This 31 days is intended to assure: 1) that the sample taken is not more than 31 days old at the time of adding the fuel oil to the storage tank, and 2) that the results of a new fuel oil sample (sample obtained prior to addition but not more than 31 days prior to) are obtained within 31 days after addition.

INSERT B48B

To minimize this potential, River Bend Station adds an antioxidant type diesel fuel oil stabilizer to new fuel in accordance with manufacturer's recommendations.

INSERT B48C

For DG 1A and 1B either the forward or rear air start subsystem is capable of starting the respective DG when air reciever pressure is within limits. For DG 1C both forward and rear air start subsystem pressures must be within limits to start the DG.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.6 (continued)

amperes and duration ensure that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

P30

~~This SR is modified by two Notes. The reason for Note 1 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. Note 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy the Surveillance.~~

SR 3.8.4.7

A battery service test is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests not to exceed 18 months.

C4

This SR is modified by ~~three~~ two Notes. Note 1 allows the once per 60 months performance of SR 3.8.4.8 in lieu of SR 3.8.4.7. This substitution is acceptable because SR 3.8.4.8 represents a more severe test of battery capacity than SR 3.8.4.7. The reason for Note 2 is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. ~~Note 3 is added to this SR to acknowledge that~~ credit may be taken for unplanned events that satisfy the Surveillance.

C4

3.8 #76 INSERT B 574

(continued)

INSERT B57A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8

A battery performance test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

3.8 #75

C11

~~and capacity is < 100% of the manufacturers rating,~~ the Surveillance Frequency is reduced to ~~12~~ 18 months. ~~However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturers rating.~~

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 8) and IEEE-485 (Ref. 11). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. ~~or every 12 months~~ If the battery shows degradation, ~~or has reached 85% of its expected life~~ if the battery. Degradation is indicated, according to IEEE-450 (Ref. 8), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. ~~All~~ These Frequencies are ~~consistent with~~ based on the recommendations in IEEE-450 (Ref. 8).

This SR is modified by ~~two~~ a Notes. The reason for the Note ~~1~~ is that performing the Surveillance would remove a required DC electrical power subsystem from service, perturb the electrical distribution system, and challenge safety systems. ~~Note 2 is added to this SR to acknowledge that credit may be taken for unplanned events that satisfy the Surveillance.~~ INSERT B58A 3.8 #76

C4

C2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE Standard 308, 1978.
4. UFSAR, Section ~~[~~8.3.2~~]~~.
5. UFSAR, Chapter ~~[~~6~~]~~.

P10 B1

(continued)

LAR 93.1481

INSERT B58A

3.8 #76

Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources—Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources—Operating."

APPLICABLE SAFETY ANALYSES P10

The initial conditions of Design Basis Accident and transient analyses in the ~~F~~USAR, Chapter ~~[6]~~ (Ref. 1) and Chapter ~~[15]~~ (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation. B1

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

C17 The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of irradiated fuel assemblies in the primary containment or Fuel Building, ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

3.8 #98

LCO C20

One ~~The~~ DC electrical power ~~subsystems, each~~ subsystem consisting of One ~~[two]~~ battery, ~~banks~~, [one ~~or two~~] battery charger~~[s]~~, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, ~~are required to be OPERABLE to~~ INSERT R00A C28

(continued)

3.8 #98

LAR 93-1481

INSERT B60A

3.8 #98

associated with Division I or II onsite Class 1E DC electrical power distribution subsystem(s) required by LCO 3.8.10 "Distribution Systems-Shutdown" is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) system is required to be OPERABLE, the Division III DC electrical power subsystem associated with the Division III onsite Class 1E DC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division I or II onsite Class 1E DC electrical power distribution subsystem(s), when portions of both Division I and II DC electrical power distribution subsystems are required to be OPERABLE by LCO 3.8.10.

3.8 #98

BASES

C 38 Shutdown

LCO (continued)

support required divisions of Distribution System divisions required OPERABLE by LCO 3.8.10, "Distribution Systems—~~Operating~~." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

APPLICABILITY

B1

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of irradiated fuel assemblies in the [primary [redacted] containment] or Fuel Building provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

INSERT B61A

3.8 #67

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.10, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC power source(s) inoperable,

(continued)

LAR 93-14R1

INSERT B61A

3.8 #67

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

LAX 93-1481

INSERT B62A

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required, ~~during periods when the DC source is required to be OPERABLE~~.

3.8
#49

ACTIONS

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

this time is acceptable for operation prior to declaring the DC batteries inoperable.

B.1

C12

When any battery parameter is outside the Category C limit ~~Allowable Value~~ for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, also are cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

3.8 #93 · 3.8 #81

The SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections ~~(at least one per month)~~ including float voltage, specific gravity, and electrolyte ~~temperature~~ Level of pilot cells. The 7 day Frequency ensures that these inspections are performed within the frequency recommended by IEEE-450 (Ref. 3).

3.8 #85

SR 3.8.6.2

72

B1 · 144 · B1

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within ~~24~~ hours of a battery ~~discharge < [110] V or a battery~~ overcharge > ~~[150]~~ V, the battery must be demonstrated to meet Category B limits. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe ~~discharge or~~ overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such ~~discharge or~~ overcharge.

C21

~~Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450 (Ref. 3), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer's recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

3.8 #81

electrolyte level, float

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose ~~temperature,~~ voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

C28, Full size FONT

The Category A limits specified for electrolyte level are based on manufacturer's recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperature~~s and charge~~ e effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendation of IEEE-450 (Ref. 3), which states that prolonged operation of cells below 2.13 V can reduce the life expectancy of cells.

(continued)

BASES (continued)

LCO

(C28) associated

from the required Class 1E battery, or from an internal AC source via rectifier with the battery available as backup.

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABLE inverters require the AC vital bus be powered by the inverter through inverted DC voltage. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).

APPLICABILITY

(B1)

The inverters required to be OPERABLE in MODES 4 and 5 and also any time during movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment or Fuel Building provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

(3.8 #67) INSERT B76A

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two divisions are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE inverters may be

(continued)

INSERT B76A

3.8 #67

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

ACTIONS

A.1 (continued)

control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

The Condition A worst scenario is one division without AC power (i.e., no offsite power to the division and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operators' attention be focused on minimizing the potential for loss of power to the remaining division by stabilizing the unit, and on restoring power to the affected division. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because:

a. There is potential for decreased safety if the unit operators' attention is diverted from the evaluations and actions necessary to restore power to the affected division to the actions associated with taking the unit to shutdown within this time limit.

b. The potential for an event in conjunction with a single failure of a redundant component in the division with AC power. (The redundant component is verified OPERABLE in accordance with Specification 5.~~Ø~~, "Safety Function Determination Program (SFDP).") ~~7.1~~ 5.5.10 ---- 5.0 #104

P13

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 2 hours. This situation could lead to a total duration of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution could be restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This results in establishing the "time zero" at the time the

(continued)

BASES

ACTIONS

A.1 (continued)

LCO was initially not met, instead of at the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B1

3.8 #72

P11

B.1

15 distribution subsystem

With one Division ~~1~~ I or ~~2~~ II AC vital ~~bus~~ distribution subsystem inoperable, the remaining OPERABLE AC vital buse~~s~~ ~~are~~ capable of supporting the minimum safety functions necessary to shut down and maintain the unit in the safe shutdown condition. Overall reliability is reduced, however, because an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus must be restored to OPERABLE status within ~~2~~ 8 hours~~.~~

P32

C28

by powering the bus from one of the associated Class 1E AC sources

distribution subsystem

Condition B represent~~s~~ one AC vital ~~bus~~ without power; potentially both the DC source and the associated AC source nonfunctioning. In this situation, the plant is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining vital buse~~s~~, and restoring power to the affected vital bus es.

3.8 #72

This ~~2~~ 8 P32 hour limit is more conservative than Completion Times allowed for the majority of components that are without adequate vital AC power. Taking exception to LCO 3.0.2 for components without adequate AC vital power, that would have Required Action Completion Times shorter than ~~2~~ 8 hours if declared inoperable, is acceptable because of: P32

a. The potential for decreased safety when requiring a change in plant conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

(continued)

BASES

ACTIONS (B1)

B.1 (continued)

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without adequate vital AC power, while not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected division;

c. The potential for an event in conjunction with a single failure of a redundant component.

(P32) The 8 hour Completion Time takes into account the importance to safety of restoring the AC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This situation could lead to a total duration of 16 hours, since initial failure of the LCO, for restoring the vital bus distribution system. At this time, an AC division could again become inoperable, and vital bus distribution could be restored to OPERABLE. This could continue indefinitely. (P32)

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This allowance results in establishing the "time zero" at the time the LCO was initially not met, instead of at the time that Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential of failing to meet the LCO indefinitely.

C.1

(P11) (B1)

(3.8 #72) With Division ~~1~~ I or ~~2~~ II DC ~~buses in one division~~ electrical power distribution subsystem inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions

(continued)

BASES

ACTIONS

C.1 (continued)

necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 2 hours, by powering the bus from the associated battery or charger. CLB

3.8 #72

Condition C may represents one division without adequate DC power, potentially with both the battery significantly degraded and the associated charger nonfunctioning. In this situation, the plant is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for loss of power to the remaining divisions, and restoring power to the affected division.

This 2 hour limit is more conservative than Completion Times allowed for the majority of components that could be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, that would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety when requiring a change in plant conditions (i.e., requiring a shutdown) while not allowing stable operations to continue;

b. The potential for decreased safety when requiring entry into numerous applicable Conditions and Required Actions for components without DC power while not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

(continued)

LAR 93-14R1

3.8 #95

INSERT B88A

nominal

TYPE	VOLTAGE	DIVISION I*	DIVISION II*	DIVISION III*
AC Electric Power Distribution System	4160 V	1ENS*SWG1A	1ENS*SWG1B	1E22*S004
	480 V LDCs	1EJS*LDC1A 1EJS*LDC2A	1EJS*LDC1B 1EJS*LDC2B	---
	480 V MCCs	1EHS*MCC2A 2C,2E,2G,2J 2L,8A,14A 15A,16A	1EHS*MCC2B 2D,2F,2H,2K 8B,14B,15B 16B	1E22*S002
	120 V Dist. Panels	1SCV*PNL2A1 2A2,2C1,2E1 2G1,2J1,2L1 8A1,14A1 15A1,16A1	1SCV*PNL2B1 2B2,2D1,2F1 2H1,2K1,8B1 14B1,15B1 16B1	1E22*S002 PNL
AC Vital Bus Electric Power Distribution System	120 VAC	1VBS*PNL01A	1VBS*PNL01B	---
DC Electric Power Distribution System	125 V	1ENB*SWG01A	1ENB*SWG01B	---
	Dist. Panels	1ENB*PNL02A 03A,04A 1ENB*MCC1	1ENB*PNL02B 03B	1E22*S001 PNL

LAR 93-14R1

BASES (continued)

ACTIONS

INSERT B91A

3.8 #67

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, ~~during~~ movement of irradiated fuel assemblies in the [primary ~~or secondary~~] containment and Fuel Building and any activities that could result in inadvertent draining of the reactor vessel). C28 B1

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal — shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, ~~these~~ A.2.1 through A.2.4 Required Actions ~~of Condition A~~ do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, ~~the~~ A.2.5 is provided to Required Actions ~~of Condition A~~ direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS. C28

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

(continued)

LAR 93-14R1

INSERT B91A

3.8 #67

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

DISCUSSION OF CHANGES TO NUREG-1434
SECTION 3.8 - ELECTRICAL POWER SYSTEMS

PLANT SPECIFIC DIFFERENCE
(continued)

P.30 The existing licensed TS do not include this limitation on performing the charger load test "during shutdown." Obviously, if the charger is rendered inoperable to perform the test, it would result in entering a TS required shutdown prior to test completion. Control over testing in an inappropriate fashion such as this, is adequately maintained by plant personnel.

P.31 This comment number not used for this station.

P.32 There are no tie breakers in the design of the electrical power distribution system at this facility.

P.33 Since the DGs at RBS do not incorporate circuitry for modified DG starts, the ITS presentation is revised accordingly. No separate 184 day surveillance test is applicable.

P.34 This comment number not used for this station.

P.35 Plant specific limits/procedures/methods for sampling and testing diesel fuel oil are included; consistent with CTS. Editorial changes also made for enhancing the appropriate plant specific changes.

P.36 This comment number not used for this station.

P.37 The current licensing basis for the DG air start system Surveillance and ACTIONS is retained. No extended time, prior to entering the DG ACTIONS, is provided with this change.

3.8 #73

P.38 The currently approved and licensed Technical Specifications for RBS do not have these surveillance requirements. The proposed change to the NUREG standard is consistent with the existing license.

LAR 93-14R1

RIVER BEND

SECTION 3.9

ATTACHMENT 1: NOT USED
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.9 REVISED PAGES

2A: MARKUP OF ITS

2B: NOT USED

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.1	Initiate action to fully insert the control rod associated with the inoperable position indicator.	Immediately
3.9 #39	AND		
	A.2.2	Initiate action to disarm the control rod drive associated with the fully inserted control rod ~~drive~~.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify the required channel has no "full-in" indication ~~on~~ for each control rod that is not "full-in." C16	Each time the control rod is withdrawn from the "full-in" position

LAR 93-14R1

INSERT 11A

REQUIRED ACTION	COMPLETION TIME
AND B.3 Initiate action to close one door in each primary containment air lock~~, except during entry and exit under administrative control~~.	Immediately

3.9 #38

-----NOTE-----
Entry and exit is permissible under administrative control.

LAR 93-1481

INSERT 13A

REQUIRED ACTION		COMPLETION TIME
AND		
B.2	Initiate action to close one door in each primary containment air lock~~, except during entry and exit under administrative control~~.	Immediately

39
#38

----NOTE------
Entry and exit is permissible under administrative control.

BASES

ACTIONS

C.1 and C.2 (continued)

sub

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling system), the reactor coolant temperature must be periodically monitored to ensure proper functioning of the alternate method. The once per hour Completion Time is deemed appropriate.

C16

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

3.9 #39 shutdown cooling

This Surveillance demonstrates that the RHR shutdown cooling subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES

None.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.9.1

3.9 #39 --- shutdown cooling

This Surveillance demonstrates that one RHR shutdown cooling subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES

None.

LAR 93-14R1

RIVER BEND

SECTION 3.10

ATTACHMENT 1: NOT USED
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

SECTION 3.10 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

LAR 93-14R1

3.10 SPECIAL OPERATIONS

3.10.7 Control Rod Testing—Operating

P10

9

3.10 #43

P9

LCO 3.10.7 The requirements of LCO 3.1.6, "Rod Pattern Control," may be suspended and control rods bypassed in the Rod Action Control System as allowed by SR 3.3.2.1.9, to allow performance of SDM demonstrations, control rod scram time testing, control rod friction testing, and ~~the Startup Test Program,~~ provided conformance to the approved control rod sequence for the specified test is verified by a second licensed operator or other qualified member of the technical staff.

APPLICABILITY: MODES 1 and 2 with LCO 3.1.6 not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Suspend performance of the test and exception to LCO 3.1.6.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.7.1	Verify movement of control rods is in compliance with the approved control rod sequence for the specified test by a second licensed operator or other qualified member of the technical staff.	During control rod movement

LAK 93-14R1

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test—Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

INSERT 19A

C1

~~a.~~ b 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 1.b of Table 3.3.2.1-1,

<u>OR</u>

2. Conformance tc the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

~~b.~~ c Each withdrawn control rod shall be coupled to the associated CRD;

B1

C5

~~c.~~ d All control rod withdrawals ~~[during out of sequence control rod moves]~~ shall be ~~made~~ in single notch ~~out~~ withdrawal mode; ~~and~~

~~d.~~ e No other CORE ALTERATIONS are in progress~~.~~; and

C22

f. CRD charging water header pressure ≥ 1520 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
INSERT 19B ~~A.~~ B. One or more of the above requirements not met ~~.~~ for reasons other than Condition A.	~~A.~~1 B Place the reactor mode switch in the shutdown or refuel position.	Immediately

C2

3.10 #47

INSERT 19A for proposed TS 3.10.8

a. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 2 requirements for Function 2.a and 2.d of Table 3.3.1.1-1;

INSERT 19B for proposed TS 3.10.8

CONDITION	REQUIRED ACTION	COMPLETION TIME
--------NOTE-------- Separate Condition entry is allowed for each control rod. -------------------- A. One control rod not coupled to its associated CRD.	A.1 ------NOTE------- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9, if required, to allow insertion of inoperable control rod and continued operation. ---------------- Fully insert inoperable control rod.	3.10 #45 3 hours
	AND	
	A.2 Disarm the associated CRD.	4 hours

BASES (continued)

ACTIONS

A.1, A.2, A.3.1, and A.3.2

These Required Actions are provided to restore compliance with the Technical Specifications overridden by this Special Operations LCO. Restoring compliance will also result in exiting the Applicability of this Special Operations LCO.

except control rod insertion

3.10 #47

All CORE ALTERATIONS, if in progress, are immediately suspended in accordance with Required Action A.1, and all insertable control rods in core cells that contain one or more fuel assemblies are fully inserted within 1 hour, in accordance with Required Action A.2. This will preclude potential mechanisms that could lead to criticality. Suspension of CORE ALTERATIONS shall not preclude the completion of movement of a component to a safe condition. Placing the reactor mode switch in the shutdown position will ensure that all inserted control rods remain inserted and result in operation in accordance with Table 1.1-1. Alternatively, if in MODE 5, the reactor mode switch ~~must~~ may be placed in the refuel position, which will also result in operating in accordance with Table 1.1-1. A Note is added to Required Action A.3.2 to indicate that this Required Action is not applicable in MODES 3 and 4, since only the shutdown position is allowed in these MODES. The allowed Completion Time of 1 hour for Required Actions A.2, A.3.1, and A.3.2 provides sufficient time to normally insert the control rods and place the reactor mode switch in the required position, based on operating experience, and is acceptable given that all operations that could increase core reactivity have been suspended.

3.10 #47

SURVEILLANCE REQUIREMENTS

SR 3.10.2.1 and SR 3.10.2.2

Meeting the requirements of this Special Operations LCO maintains operation consistent with or conservative to operating with the reactor mode switch in the shutdown position (or the refuel position for MODE 5). The functions of the reactor mode switch interlocks that are not in effect, due to the testing in progress, are adequately compensated for by the Special Operations LCO requirements. The administrative controls are to be periodically verified to ensure that the operational requirements continue to be met. The Surveillances performed at the 12 hour and 24 hour Frequencies are intended to provide appropriate assurance

(continued)

LAR 93-14R1

B 3.10 SPECIAL OPERATIONS

B 3.10.3 Single Control Rod Withdrawal—Hot Shutdown

BASES

BACKGROUND	The purpose of this MODE 3 Special Operations LCO is to permit the withdrawal of a single control rod for testing while in hot shutdown, by imposing certain restrictions. In MODE 3, the reactor mode switch is in the shutdown position, and all control rods are inserted and blocked from withdrawal. Many systems and functions are not required in these conditions, due to other installed interlocks that are actuated when the reactor mode switch is in the shutdown position. However, circumstances ~~will~~ may arise while in MODE 3 that present the need to withdraw a single control rod for various tests (e.g., friction tests, scram timing, and coupling integrity checks). These single control rod withdrawals are normally accomplished by selecting the refuel position for the reactor mode switch. This Special Operations LCO provides the appropriate additional controls to allow a single control rod withdrawal in MODE 3.
C19	
APPLICABLE SAFETY ANALYSES	With the reactor mode switch in the refuel position, the analyses for control rod withdrawal during refueling are applicable and, provided the assumptions of these analyses are satisfied in MODE 3, these analyses will bound the consequences of an accident. Explicit safety analyses in the ~~U~~SAR (Ref. 1) demonstrate that the functioning of the refueling interlocks and adequate SDM will preclude unacceptable reactivity excursions.
P3	
	Refueling interlocks restrict the movement of control rods to reinforce operational procedures that prevent the reactor from becoming critical. These interlocks prevent the withdrawal of more than one control rod. Under these conditions, since only one control rod can be withdrawn, the core will always be shut down even with the highest worth control rod withdrawn if adequate SDM exists.
3.10 #23	The control rod scram function provides backup protection ~~to~~ in the event normal refueling procedures and the refueling interlocks ~~which~~ fail to prevent inadvertent criticalities during refueling.
C20	

(continued)

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

a five by five array of control rods, centered on the withdrawn control rod, are inserted and incapable of withdrawal. This alternate backup protection is required when removing the CRD because this removal renders the withdrawn control rod incapable of being scrammed.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

3.10 #48

C18

C6

~~LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operations," and~~

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation in MODE 4 with the reactor mode switch in the refuel position can be performed in accordance with ~~other LCOs (i.e.,~~ Special Operations LCO 3.10.2, "Reactor Mode Switch Interlock Testing," ~~and LCO 3.10.3, "Single Control Rod Withdrawal—Hot Shutdown")~~ without meeting this Special Operations LCO or its ACTIONS. If a single control rod withdrawal is desired in MODE 4, controls consistent with those required during refueling must be implemented and this Special Operations LCO applied. INSERT B16A

C8

The refueling interlocks of LCO 3.9.2, "Refuel Position One-Rod-Out Interlock," required by this Special Operations LCO ~~3.10.4~~ will ensure that only one control rod can be withdrawn. At the time CRD removal begins, the disconnection of the position indication probe will cause LCO 3.9.4, "Control Rod Position Indication," and therefore, LCO 3.9.2 to fail ~~to be met. At this time,~~ Therefore, prior to commencing CRD removal, a control rod withdrawal block ~~will~~ is required to be inserted to ensure that no additional control rods can be withdrawn and that compliance with this Special Operations LCO is maintained.

C14

C3

To back up the refueling interlocks (LCO 3.9.2) or the control rod withdrawal block, the ability to scram the withdrawn control rod in the event of an inadvertent criticality is provided by ~~this~~ the Special Operations LCO requirements in Item c.1. Alternatively, when the scram function is not OPERABLE, or the CRD is to be removed, a sufficient number of rods in the vicinity of the withdrawn control rod are required to be inserted and made incapable

(continued)

LAR 93-14R1

BASES

C8 (Item C.2) INSERT B17A C7

LCO (continued)

of withdrawal. This precludes the possibility of criticality upon withdrawal of this control rod.

APPLICABILITY

Control rod withdrawals are adequately controlled in MODES 1, 2, and 5 by existing LCOs. In MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.3, or this Special Operations LCO, and if limited to one control rod. This allowance is only provided with the reactor mode switch in the refuel position.

During these conditions, the full insertion requirements for all other control rods, the one-rod-out interlock (LCO 3.9.2), control rod position indication (LCO 3.9.4), and scram functions (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and LCO 3.9.5, "Control Rod OPERABILITY—Refueling"), or the added administrative controls in Item b.2 and Item c.2 of this Special Operations LCO, provide mitigation of potential reactivity excursions.

3.10 #48

ACTIONS

C5 divisions 4

A Note has been provided to modify the ACTIONS related to a single control rod withdrawal while in MODE ~~3~~. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent ~~trains~~, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for each requirement of the LCO not met provide appropriate compensatory measures for separate requirements that are not met. As such, a Note has been provided that allows separate Condition entry for each requirement of the LCO.

C8

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

If one or more of the requirements of this Special Operations LCO are not met with the affected control rod insertable, these Required Actions restore operation consistent with normal MODE 4 conditions (i.e., all rods

(continued)

ACTIONS

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

C8 's

3.10 #27

inserted) or with the exceptions allowed in this Special Operations LCO. Required Action A.1 has been modified by a Note that clarifies ~~that~~ 's the intent of any other LCO Required Action~~s, in accordance with the other applicable LCOs,~~ to insert all control rods. This Required Action includes exiting this Special Operations Applicability LCO by returning the reactor mode switch to the shutdown position. A second Note has been added to Required Action A.1 to clarify that this Required Action is only applicable if the requirements not met are for an affected LCO.

C8

Required Actions A.2.1 and A.2.2 are specified, based on the assumption that the control rod is being withdrawn. If the control rod is still insertable, actions must be immediately initiated to fully insert all insertable control rods and within 1 hour place the reactor mode switch in the shutdown position. Action must continue until all such control rods are fully inserted. The allowed Completion Time of 1 hour for placing the reactor mode switch in the shutdown position provides sufficient time to normally insert the control rods.

B.1, B.2.1, and B.2.2

If one or more of the requirements of this Special Operations LCO are not met with the affected control rod not insertable, withdrawal of the control rod and removal of the associated CRD must immediately be suspended. If the CRD has been removed, such that the control rod is not insertable, the Required Actions require the most expeditious action be taken to either initiate action to restore the CRD and insert its control rod, or restore compliance with this Special Operations LCO.

SURVEILLANCE REQUIREMENTS

SR 3.10.4.1, SR 3.10.4.2, SR 3.10.4.3, and SR 3.10.4.4

The other LCOs made applicable by this Special Operations LCO are required to have their associated Surveillances met to establish that this Special Operations LCO is being met. If the local array of control rods is inserted and disarmed while the scram function for the withdrawn rod is not

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.5 Single Control Rod Drive (CRD) Removal—Refueling

BASES

BACKGROUND

The purpose of this MODE 5 Special Operations LCO is to permit the removal of a single CRD during refueling operations by imposing certain administrative controls. Refueling interlocks restrict the movement of control rods and the operation of the refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn from a core cell containing one or more fuel assemblies. The refueling interlocks use the "full in" position indicators to determine the position of all control rods. If the "full in" position signal is not present for every control rod, then the all rods in permissive for the refueling equipment interlocks is not present and fuel loading is prevented. Also, the refuel position one-rod-out interlock will not allow the withdrawal of a second control rod.

C20

The control rod scram function and provides backup protection ~~to~~ in the event normal refueling procedures ~~as do~~ the refueling interlocks described above ~~which~~ fail to prevent inadvertent criticalities during refueling. The requirement for this function to be OPERABLE precludes the possibility of removing the CRD once a control rod is withdrawn from a core cell containing one or more fuel assemblies. This Special Operations LCO provides controls sufficient to ensure the possibility of an inadvertent criticality is precluded, while allowing a single CRD to be removed from a core cell containing one or more fuel assemblies. The removal of the CRD involves disconnecting the position indication probe, which causes noncompliance with LCO 3.9.4, "Control Rod Position Indication," and, therefore, LCO 3.9.1, "Refueling Equipment Interlocks," and LCO 3.9.2, "Refueling Position One-Rod-Out Interlock." The CRD removal also requires isolation of the CRD from the CRD Hydraulic System, thereby causing inoperability of the control rod (LCO 3.9.5, "Control Rod OPERABILITY—Refueling").

3.10 #28

C8

(continued)

BASES (continued)

3.10 #48

ACTIONS

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

LCO 3.3.8.2,

If one or more of the requirements of this Special Operations LCO are not met, the immediate implementation of these Required Actions restores operation consistent with the normal requirements for failure to meet LCO 3.3.1.1, LCO 3.9.1, LCO 3.9.2, LCO 3.9.4, and LCO 3.9.5 (i.e., all control rods inserted) or with the allowances of this Special Operations LCO. The Completion Times for Required Action A.1, Required Action A.2.1, and Required Action A.2.2 are intended to require these Required Actions be implemented in a very short time and carried through in an expeditious manner to either initiate action to restore the CRD and insert its control rod, or initiate action to restore compliance with this Special Operations LCO. Actions must continue until either Required Action A.2.1 or Required Action A.2.2 is satisfied.

SURVEILLANCE REQUIREMENTS

SR 3.10.5.1, SR 3.10.5.2, SR 3.10.5.3, SR 3.10.5.4, and SR 3.10.5.5

Verification that all the control rods, other than the control rod withdrawn for the removal of the associated CRD, are fully inserted is required to ensure the SDM is within limits. Verification that the local five by five array of control rods other than the control rod withdrawn for the removal of the associated CRD, is inserted and disarmed, while the scram function for the withdrawn rod is not available, is required to ensure that the possibility of criticality remains precluded. Verification that a control rod withdrawal block has been inserted ensures that no other control rods can be inadvertently withdrawn under conditions when position indication instrumentation is inoperable for the withdrawn control rod. The Surveillance for LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," which is made applicable by this Special Operations LCO, is required in order to establish that this Special Operations LCO is being met. Verification that no other CORE ALTERATIONS are being made is required to ensure the assumptions of the safety analysis are satisfied.

C8

Periodic verification of the administrative controls established by this Special Operations LCO is prudent to preclude the possibility of an inadvertent criticality. The 24 hour Frequency is acceptable, given the administrative

(continued)

LAR 93-14R1

BASES

APPLICABLE SAFETY ANALYSES (continued)

the cell. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain inserted. Prior to reloading fuel into the cell, however, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur, as evaluated in the Reference 1 analysis.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation in MODE 5 with LCO 3.9.3, "Control Rod Position," LCO 3.9.4, "Control Rod Position Indication," or LCO 3.9.5, "Control Rod OPERABILITY—Refueling," not met, can be performed in accordance with the Required Actions of these LCOs without meeting this Special Operations LCO or its ACTIONS. If multiple control rod withdrawal or removal, or CRD removal is desired, all four fuel assemblies are required to be removed from the associated cells. Prior to entering this LCO, any fuel remaining in a cell whose ~~control rod~~ CRD was previously removed under the provisions of another LCO must be removed. INSERT B26A INSERT B26B

C9 C6 C15

When loading fuel into the core with multiple control rods withdrawn, special spiral reload sequences are used to ensure that reactivity additions are minimized. Otherwise, all control rods must be fully inserted before loading fuel.

3.10 #31

P16 (B.A.1) del

APPLICABILITY

Operation in MODE 5 is controlled by existing LCOs. The exceptions from other LCO requirements (e.g., the ACTIONS of LCO 3.9.3, LCO 3.9.4 or LCO 3.9.5) allowed by this Special Operations LCO are appropriately controlled by requiring all fuel to be removed from cells whose "full in" indicators are allowed to be bypassed.

(continued)

BASES (continued) A.2 C8

ACTIONS C10 A.1, A.3.1, and A.3.2

Commences activities which will

If one or more of the requirements of this Special Operations LCO are not met, the immediate implementation of these Required Actions ~~restores~~ operation consistent with the normal requirements for refueling (i.e., all control rods inserted in core cells containing one or more fuel assemblies) or with the exceptions granted by this Special Operations LCO. The Completion Times ~~for Required Action A.1, Required Action A.2.1, and Required Action A.2.2~~ are intended to require that these Required Actions be implemented in a very short time and carried through in an expeditious manner~~, to either initiate action to restore the affected CRDs and insert their control rods, or initiate action to restore compliance with this Special Operations LCO.~~

C10

SURVEILLANCE REQUIREMENTS

SR 3.10.6.1, SR 3.10.6.2, and SR 3.10.6.3

Periodic verification of the administrative controls established by this Special Operations LCO is prudent to preclude the possibility of an inadvertent criticality. The 24 hour Frequency is acceptable, given the administrative controls on fuel assembly and control rod removal, and takes into account other indications of control rod status available in the control room.

REFERENCES P3

1. UFSAR, Section 15.4.1.1. B1

3.10 #31 2. NSAC-164L, "Guidelines for BWR Reactivity Control During Refueling," April 1992.

P116

B 3.10 Special Operations

B 3.10.7 Control Rod Testing—Operating

BASES

BACKGROUND

3.1 #43

P9

The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod pattern controller (RPC) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RPC. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests may include SDM demonstrations, control rod scram time testing, ~~control rod friction testing,~~ and ~~testing performed during the Startup Test Program.~~ This Special Operations LCO provides the necessary exceptions to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.

APPLICABLE SAFETY ANALYSES

C3

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RPC provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analyses are not violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analyses of References ~~1, 2,~~ 2, and 3 may not be preserved. Therefore, special CRDA analyses are required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

(continued)

LAR 93-14R1

BASES

3.10 #43

LCO (continued)

not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control" (i.e., out of sequence control rod withdrawals) must be made in the C5 single notch withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. INSERT B34A C22 This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

ACTIONS C2

A.1 and A.2 INSERT B34B

With one or more of the requirements of this LCO not met for reasons other than an uncoupled control rod, the testing should be immediately stopped by placing the reactor mode switch in the shutdown or refuel position. This results in a condition that is consistent with the requirements for MODE 5 where the provisions of this Special Operations LCO are no longer required.

SURVEILLANCE REQUIREMENTS C1

SR 3.10.8.1, SR 3.10.8.2 and SR 3.10.8.3 INSERT B34C

~~The~~ control rod withdrawal sequences during the SDM tests may be enforced by the RPC (LCO 3.3.2.1, Function 1b, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff. As noted, either the applicable SRs for the RPC (LCO 3.3.2.1) must be

(continued)

LAR 93-1481

INSERT B34A

Furthermore, since the control rod scram function with the RCS at atmospheric pressure relies solely on the CRD accumulator, it is essential that the CRD charging water header remain pressurized.

INSERT B34B

If a control rod is discovered to be uncoupled during this Special Operation, a controlled insertion of the uncoupled control rod is required; either to attempt recoupling, or to preclude a control rod drop. This controlled insertion is preferred since if the control rod fails to follow the drive as it is withdrawn (i.e., is "stuck" in an inserted position), placing the reactor mode switch in the shutdown position per Required Action B.1 could cause substantial secondary damage. If recoupling is not accomplished, operation may continue, provided the control rods ~~are~~ fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. Required Actions A.1 ~~is~~ modified by a Note that allows control rods to be bypassed in the Rod Action Control System (RACS) if required to allow insertion of the inoperable control rods and continued operation. SR 3.3.2.1.9 provides additional requirements when the control rods are bypassed to ensure compliance with the CRDA analysis.

3.10 #45

and A.2 are

and disarming

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

Condition A is modified by a Note allowing separate Condition entry for each uncoupled control rod. This is acceptable since the Required Actions for this Condition provide appropriate compensatory actions for each uncoupled control rod. Complying with the Required Actions may allow for continued operation. Subsequent uncoupled control rods are governed by subsequent entry into the Condition and application of the Required Actions.

B.1

. . .

LAR 93-14R1

INSERT B35A

SR 3.10.8.6

3.10 #40 1

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 1520 psig is well below the expected pressure of 1750 psig ~~to 2000 psig~~. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

(LAR 93-1481)

DISCUSSION OF CHANGES TO NUREG 1434
SECTION 3.10 - SPECIAL OPERATIONS

PLANT SPECIFIC DIFFERENCE
(continued)

(3.10 #31)

~~P.16 An additional reference is provided for clarity.~~

CHANGE/IMPROV MENT TO NUREG STS

C.1 Amendments 41 and 7 to Limerick Generating Station Units 1 and 2 (NPF-39 and NPF-85), respectively, issued July 30, 1990, eliminated APRM RPS trip OPERABILITY requirements during MODE 5, other than during SDM demonstrations. This remaining requirement is therefore moved into this SHUTDOWN MARGIN demonstration Special Operation TS. The SER for these amendments outlined various levels of control to prevent inadvertent reactor criticality and fuel damage during refueling operations. These are equally applicable to BWRs in general, and include:

1. Licensed plant operators are trained to operate equipment and follow approved procedures.

2. Plant approved refueling and maintenance procedures specify core alteration steps.

3. SRMs indicate the potential for reactor criticality and generate a control rod block signal on high neutron flux levels.

4. Refueling interlocks prevent the removal of more than one control rod and prevent the insertion of fuel bundles into the core unless all control rods are fully inserted.

5. The IRMs provide an indication of local power. IRMs will provide control rod blocks and scram signals on high neutron flux levels.

The NRC Staff agreed in the SER with the conclusion presented in the amendment request for Limerick, which is equally applicable to BWRs in general, that the APRMs are not necessary for safe operation of the plant while operating in MODE 5 with the mode switch in "Refuel" for the following reasons:

- The IRMs are a safety related subsystem of the NMS and are required by TS to be OPERABLE in MODE 5 (with a control rod withdrawn). The IRMs will generate an RPS scram or control rod block if neutron flux increased to the applicable setpoint.

- The IRMs and SRMs are designed and calibrated to be more sensitive to neutron flux than the APRMs.

LAR 93-14R1

RIVER BEND

CHAPTER 4

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

CHAPTER 4 REVISED PAGES

1A: MARKUP OF CTS

1B: NOT USED

1C: NOT USED

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-1421

4.0 #7

4.0 ~~5.0~~ DESIGN FEATURES

4.1 ~~5.1~~ SITE

Location: ~~and the~~ exclusion area boundary shall have a radius of 3000 ft from the centerline of the reactor.

Site and EXCLUSION AREA Boundaries

4.1.1 The ~~exclusion~~ site boundary area shall ~~be as shown in Figure 5.1.1-1~~ include the area

LOW POPULATION ZONE (LPZ)

4.1.2 The low population zone shall ~~be as shown in Figure 5.1.2-1~~ have a radius of 2.5 miles from the centerline of the reactor.

~~MAPS DEFINING UNRESTRICTED AREAS AND~~ SITE BOUNDARY ~~FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS~~

MOVE TO 4.1.1

5.1.3 Information regarding radioactive gaseous and liquid effluents which will allow 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 as shown in Figures 5.1.1-1 and 5.1.3-1.

A1 A2 LA1 A2

5.2 CONTAINMENT

PRIMARY CONTAINMENT

5.2.1 The primary containment is a steel structure composed of a vertical right cylinder and a torispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head. Primary containment contains approximately 20 feet of water in the suppression pool which is connected to the drywell through a series of horizontal vents. The primary containment has a minimum net free air volume of 1,190,000 cubic feet. The drywell has a minimum net free air volume of 236,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment and drywell are designed and shall be maintained for:

LA2

a. Maximum internal pressure:
   1. Drywell 25 psig.
   2. Primary Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external-to-internal differential pressure:
   1. Drywell 20 psid.
   2. Primary Containment 0.6 psid.

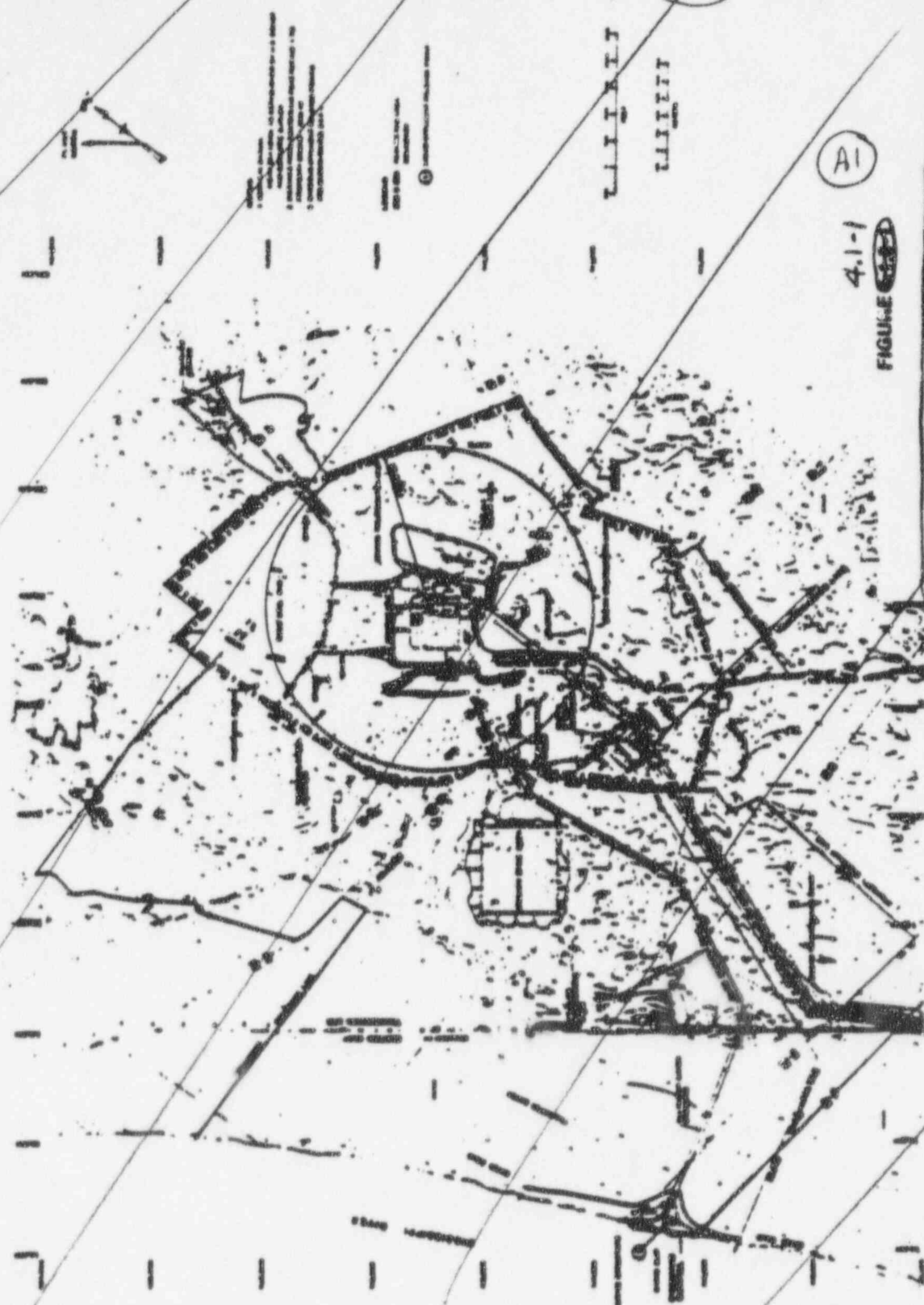
SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the shield building, the auxiliary building and the fuel building. Secondary containment has a minimum free volume of 2,249,400 cubic feet.

LAR 95-14R1

4.0 #7

{Omit EAB and other unnecessary detail.} LA3



A1

FIGURE ~~5.1-1~~ 4.1-1 ~~MAP DEFINING UNRESTRICTED AREAS AND~~ SITE BOUNDARY ~~FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS~~

e

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

CHAPTER 4 REVISED PAGES

2A: MARKUP OF ITS

2B: NOT USED

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

The River Bend Station is located in West Feliciana Parish, Louisiana, on the east bank of the Mississippi River approximately 24 miles north-northwest of Baton Rouge (city center), Louisiana. The site comprises approximately 3,342 acres.

4.0 DESIGN FEATURES

4.1 Site Location

4.0 #7

B1 ~~4.1.1 Site and Exclusion Area Boundaries~~

~~The site and exclusion area boundaries [shall be as described or as shown in Figure 4.1-1].~~ The exclusion area boundary shall have a radius of 3000 ft from the centerline of the reactor.

B1 ~~4.1.2 Low Population Zone (LPZ)~~

The LPZ ~~[shall be as described or as shown in Figure 4.1-2]~~ shall have a radius of 2.5 miles from the centerline of the reactor.

4.2 Reactor Core

4.2.1 Fuel Assemblies

B1 C1 B4

The reactor shall contain ~~[800]~~ 624 fuel assemblies. Each assembly shall consist of a matrix of ~~zirconium alloy~~ Zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material~~[, and water rods]~~. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

B1

The reactor core shall contain ~~[193]~~ 145 cruciform shaped control rod assemblies. The control material shall be ~~[~~boron carbide~~,~~ or hafnium metal~~]~~ ~~as approved by the NRC~~, or both.

C2

4.0
#7

INSERT 4A B1

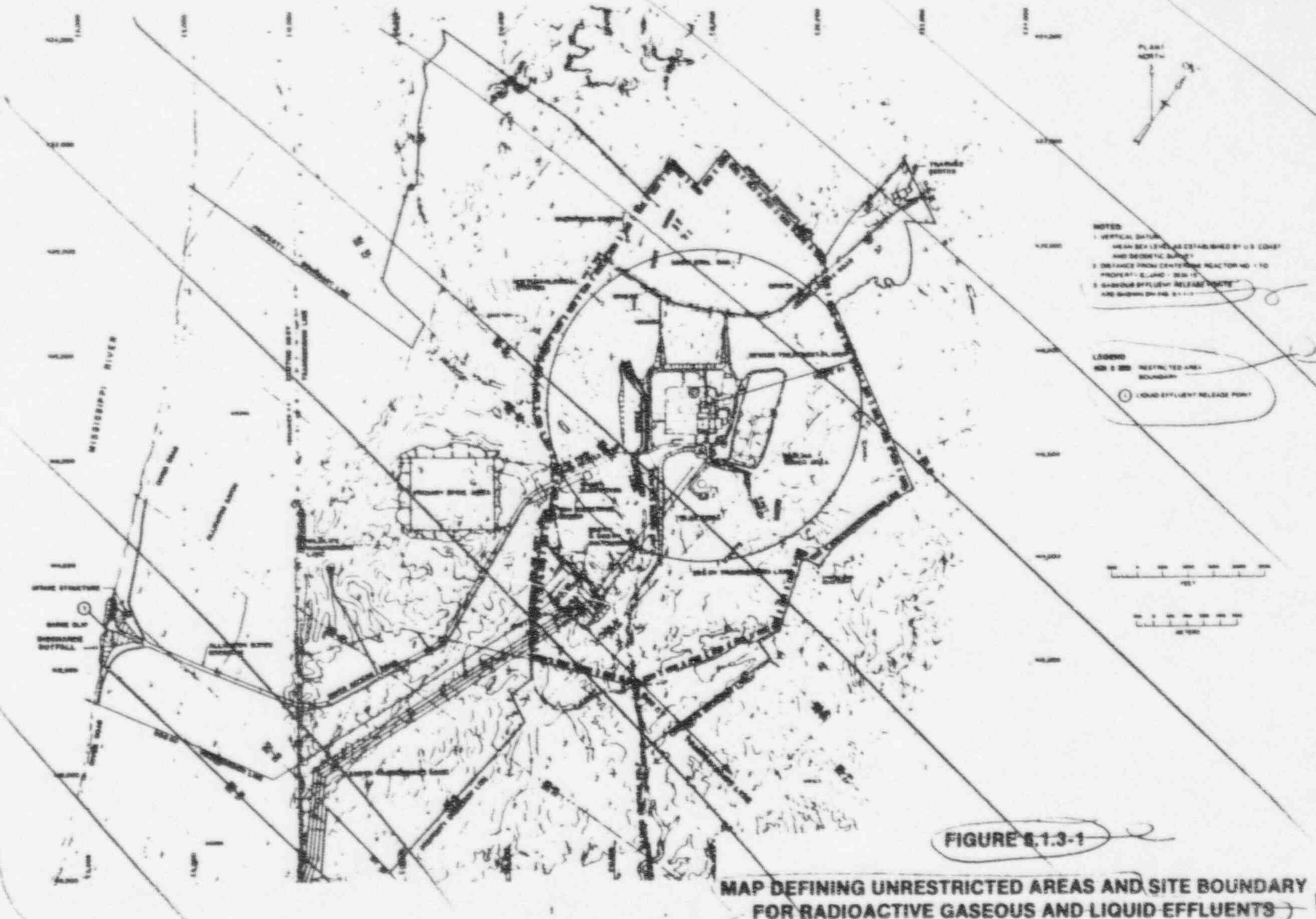
This figure shall consist of [a map of] the site area and provide, as a minimum, the information described in Section [2.1.2] of the FSAR relating to [the map].

Figure 4.1-1 (page 1 of 1)
Site ~~and Exclusion~~ Area Boundar~~ies~~y

RIVER BEND - UNIT 1

5-4

INSERT 4A



4.0 #7

FIGURE 5.1.3-1

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

LAR 93-14R1

LAR 93-14R1

RIVER BEND

CHAPTER 5

ATTACHMENT 1: CTS-PSTS COMPARISON DOCUMENT
ATTACHMENT 2: ITS-PSTS COMPARISON DOCUMENT

LAR 93-14R1

ATTACHMENT 1

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

CHAPTER 5 REVISED PAGES

1A: MARKUP OF CTS

1B: DISCUSSION OF CHANGES

1C: NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14R1

ATTACHMENT 1A

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF CTS

LAR 93-14R1

 ⟨5.1 & 5.2⟩

5.1
~~6.1~~ RESPONSIBILITY

6.0 #2

5.1.1
~~6.1.1~~ The Plant Manager (A1) shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

LA1

5.1.2
~~6.1.2~~ The Shift Supervisor (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 Senior Vice President - River Bend Nuclear Group, shall be reissued to all station personnel on an annual basis.~~ A8

5.2
~~6.2~~ ORGANIZATION

5.2.1
~~6.2.1~~ OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

5.2.1.a ~~a.~~ Lines of authority, responsibility, and communication shall be established and defined for 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 organization 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 Chapter 13 of the Updated Safety Analysis Report.

5.2.1.b ~~b.~~ The Plant Manager (A4) shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

5.2.1.c ~~c.~~ The Senior Vice President - RBNG (A4) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The specified corporate executive shall be documented in the USAR. (A4)

5.2.1.d ~~d.~~ The individuals who train the operating staff and those who carry out health physics 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
~~6.2.2~~ UNIT STAFF

LA3 a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

INSERT 1A →

LAR 93-14R1

INSERT 1A

A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site while the unit is in MODE 1, 2, or 3.

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6.0 ADMINISTRATIVE CONTROLS

UNIT STAFF (Continued)

5.2.2.b ~~b.~~ At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room;

A4

5.2.2.d ~~c.~~ A Radiation Protection Technician* shall be on site when fuel is in the reactor;

A10

d. ~~All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;~~

A6

LA7

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, the Control Operating Foreman, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

5.0 #6

LA7

A4 A6

5.2.2.d *The Radiation Protection Technician and fire brigade composition 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.

LAR 93-14R1

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UNIT STAFF (Continued)

5.2.2e ~~f.~~ Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel).

5.0 #4

A2

LA9

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 42-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

A4

Any deviation from the above guidelines shall be authorized by the Plant Manager or either one of the Assistant Plant Managers or the Director-Radiological Programs, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

5.2.2.f ~~g.~~ The Operations Supervisor, Assistant Operations Supervisor, Shift Supervisor and the Control Operating Foreman, as specified in Chapter 13 of the Updated Safety Analysis Report, shall hold a senior reactor operator license. The Nuclear Control Operators shall hold, as a minimum, a reactor operator license as specified in Chapter 13 of the Updated Safety Analysis Report.

LA1

LAR 92-14R1

FIGURE 6.2.1-1
This figure has been deleted

LAR 93-14R1

FIGURE 6.2.2-1
This figure has been deleted

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA*	1	None

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⟨5.2.2.a⟩ A4

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TABLE NOTATION

SS - Shift Supervisor with a Senior Operator license on Unit 1.
SRO - Individual with a Senior Operator license on Unit 1.
RO - Individual with an Operator license on Unit 1.
AO - Auxiliary operator
STA - Shift Technical Advisor

LA3

A5 *The Shift Technical Advisor (STA) position may be filled by an on-shift Shift Supervisor (SS) or Senior Reactor Operator (SRO) provided the individual meets the STA qualifications of Specification 6.2.4 and five (5) licensed operators are on shift.

5.2.2.c The shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 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 6.2.2-1. 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.

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5.1.2 During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

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~~6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)~~

~~FUNCTION~~

~~6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Manager-River Bend Oversight.~~

~~COMPOSITION~~

~~6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located onsite. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, of which at least 1 year experience shall be in the nuclear field.~~

~~RESPONSIBILITIES~~

~~6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.~~

~~RECORDS~~

~~6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager-River Bend Oversight.~~ LA5

~~6.2.4~~ SHIFT TECHNICAL ADVISOR

5.2.2g

~~6.2.4.1~~ The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The dedicated Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

For the dual role position shown in Table 6.2.2-1, the Shift Technical Advisor shall have a bachelor's degree in engineering, engineering technology, or physical science from an accredited institution, including course work in the mathematical or engineering sciences, or a professional engineer's license obtained by successful completion of the PE examination and shall have received all of the training for the normal STA position described above. A5

*Not responsible for sign-off function. LA5

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5.3

~~6.3~~ UNIT STAFF QUALIFICATIONS

5.3.1

~~6.3.1~~ Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the ~~Director-~~ A2 ~~Radiological Programs~~ who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall LA3 also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

~~6.4 TRAINING~~

LA1 ~~6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director - Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.~~

~~6.5 REVIEW AND AUDIT~~

~~6.5.1 FACILITY REVIEW COMMITTEE (FRC)~~

~~FUNCTION~~

~~6.5.1.1 The FRC shall function to advise the Plant Manager on all matters related to nuclear safety.~~

~~COMPOSITION~~

~~6.5.1.2 The FRC shall consist of at least six but not more than eleven members and shall be composed of:~~

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~~The Chairperson assigned by the Plant Manager. The Chairperson will be chosen from the personnel reporting directly to the Plant Manager.~~

~~The remainder of the members shall be from the personnel reporting directly to the Plant Manager and the supervisory staff. This composition will be representative of the various disciplines.~~

~~All members shall be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.4, prior to being approved by the Chairperson. This qualification will be maintained while assigned to FRC activities.~~

~~ALTERNATES~~

~~6.5.1.3 All alternate members shall be appointed in writing by the FRC Chairperson and shall be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.4, prior to being approved by the Chairperson. This qualification shall be maintained while assigned to FRC activities.~~

MEETING FREQUENCY

6.5.1.4 The FRC shall meet at least once per calendar month and as convened by the FRC Chairperson or designated alternate.

QUORUM

6.5.1.5 The Quorum of the FRC necessary for the performance of the FRC responsibility and authority provisions of these Technical Specifications shall consist of a majority of the members including the Chairperson or Alternate Chairperson, and no more than two alternate members.

RESPONSIBILITIES

6.5.1.6 The FRC shall be responsible for:

a. Review of all plant general administrative procedures and changes thereto;

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

c. Review of all proposed changes to Appendix A Technical Specifications;

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

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 Senior Vice President - RBNG and the Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety; items that may be included in this review are NRC inspection reports that require written response, QA audits/surveillance findings of operating and maintenance activities, NRB audit results, and American Nuclear Insurer (ANI) inspection results;

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

i. Review of initial start-up testing phase start-up procedures and revisions; and

j. Review of the Emergency Plan and implementation procedures at least once per 12 months and all proposed changes thereto.

6.5.1.7 The FRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6.a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6.a. through e. constitutes an unreviewed safety question.

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RESPONSIBILITIES (Continued)

c. Provide written notification within 24 hours to the Senior Vice President - RBNG and the Nuclear Review Board of disagreement between the FRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The FRC shall maintain written minutes of each FRC meeting that, at a minimum, document the results of all FRC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Plant Manager and the NRB.

6.5.2 TECHNICAL REVIEW AND CONTROL

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group that prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group that prepared the procedure. Each such procedure and program, or changes thereto, shall be approved, prior to implementation, by the Plant Manager, one of the Assistant Plant Managers or the Director - Radiological Programs, or by the manager/department head responsible for the program or the activity described in the procedure, with the exception of the Emergency Plan and implementing procedures which shall be approved by the Manager - Administration, Plant Manager and Senior Vice President - RBNG.

6.5.2.2 Individuals responsible for reviews performed in accordance with Section 6.5.2.1 shall be members of River Bend Nuclear Group supervisory staff, and the reviews shall be performed in accordance with administrative procedures. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.3 The station security program and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.4 The station emergency plan and implementing procedures and recommended changes shall be approved in accordance with Specification 6.5.2.1.

6.5.2.5 The station fire protection plan and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.6 Records documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.5 shall be maintained.

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~~6.5.3 NUCLEAR REVIEW BOARD (NRB)~~

~~FUNCTION~~

~~6.5.3.1 The NRB shall function to provide independent review and audit 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,~~
~~i. Licensing and regulatory affairs,~~
~~j. Training.~~

~~The NRB shall report to and advise the Senior Vice President - RBNG on those areas of responsibility in Specifications 6.5.3.7 and 6.5.3.8.~~

~~COMPOSITION~~

~~6.5.3.2 The NRB shall be composed of at least nine but not more than thirteen individuals who shall possess the necessary expertise to provide the independent review and audit functions identified in Specification 6.5.3.1. All members shall be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.7, prior to being approved by the Chairperson. This qualification will be maintained while assigned to NRB activities. The Senior Vice President-RBNG provides nominations for permanent NRB membership to the NRB Chairperson for review and approval.~~

~~ALTERNATES~~

~~6.5.3.3 All alternate members shall be appointed in writing by the NRB Chairperson and be qualified to the applicable portions of ANSI/ANS 3.1-1978, Section 4.7, prior to being approved by the Chairperson. This qualification will be maintained while assigned to NRB activities.~~

~~CONSULTANTS~~

~~6.5.3.4 Consultants shall be utilized as determined by the NRB Chairperson to provide expert advice to the NRB.~~

~~MEETING FREQUENCY~~

~~6.5.3.5 The NRB shall meet at least once per 6 months.~~

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QUORUM

6.5.3.6 The Quorum of the NRB necessary for the performance of the review and audit functions of these technical specifications shall consist of a majority of the members including the Chairperson or Vice Chairperson, and no more than two alternate members. No more than a minority of the quorum shall have line responsibility for the operation of the unit.

REVIEW

6.5.3.7 The NRB shall be responsible for the review of:

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 an 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 this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

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

g. All REPORTABLE EVENTS;

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; and

i. Reports and meeting minutes of the FRC.

AUDITS

6.5.3.8 Audits of unit activities shall be performed under the cognizance of the NRB. 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 unit staff, at least once per 12 months;

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LA4

AUDITS (Continued)

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 Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures, at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation, at least once per 12 months, utilizing either qualified offsite licensee personnel or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year; and

g. Any other area of unit operation considered appropriate by the NRB, Plant Manager or the Senior Vice President - RBNG.

h. The Emergency Plan and implementing procedures, at least once per 12 months.

i. The Security Plan and implementing procedures, at least once per 12 months.

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 by the Quality Assurance Program for effluent and environmental monitoring, at least once per 12 months.

RECORDS

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

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RECORDS (Continued)

a. ~~Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following each meeting.~~

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b. ~~Reports of reviews encompassed by Specification 6.5.3.7 shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following completion of the review.~~

c. ~~Audit reports encompassed by Specification 6.5.3.8 shall be forwarded to the Senior Vice President - RBNG, and to the management positions responsible for the areas audited, within 30 days after completion of the audit by the auditing organization.~~

~~6.6 REPORTABLE EVENT ACTION~~

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~~6.6.1 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~~

LA1

b. Each REPORTABLE EVENT shall be reviewed by the FRC and the results of this review shall be submitted to the NRB and the Plant Manager.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

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a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - RBNG and the NRB chairman (or personnel acting for their function) shall be notified within 24 hours.

LA1

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, LA1 the NRB, and the Senior Vice President - RBNG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

5.4 ~~6.8~~ PROCEDURES AND PROGRAMS

5.4.1 ~~6.8.1~~ Written procedures shall be established, implemented, and maintained covering the activities referenced below:

PROCEDURES AND PROGRAMS (Continued)

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

5.4.1.b ~~b.~~ The ~~applicable~~ emergency operating procedures required to implement the requirements of NUREG-0737 and supplements thereto. L2 5.0 #30

c. Refueling operations.

<5.4.1.a> d. Surveillance and test activities of safety-related equipment. A1

e. Security Plan implementation.

f. Emergency Plan implementation. A2

5.4.1.d ~~g.~~ Fire Protection Program implementation.

h. Process Control Program implementation. LA6

<5.4.1.e> i. Offsite Dose Calculation Manual implementation. A3

5.4.1.c ~~j.~~ Quality Assurance Program for effluent and environmental monitoring.

LA4

~~6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with Specification 6.5.2.1.~~

~~6.8.3 Temporary changes to procedures of Specification 6.8.1 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 Operator license on the unit affected; and~~

~~c. The change is documented, reviewed by the FRC as required by Specification 6.5.1.6, and approved in accordance with Specification 6.5.2.1 within 14 days of implementation.~~

5.5 ~~6.8.4~~ The following programs shall be established, implemented, and maintained: A4

5.5.2 ~~a.~~ Primary Coolant Sources Outside Containment

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 practicable levels. The systems include the HPCS, LPCS, RHR, RCIC, process sampling, and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

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PROCEDURES AND PROGRAMS (Continued)

2. Integrated leak test requirements for each system, at refueling cycle intervals or more frequently.

b. In-Plant Radiation Monitoring

LA7

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

5.5.3 ~~c.~~ Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents and containment atmosphere samples, under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

L4

3.0 #28

~~A9~~

d. Biofouling Prevention and Detection

A program, which will include the procedures to prevent biofouling of safety-related equipment, to assure detection of Corbicula in the intake embayment and the clarifier influent, and to monitor and survey safety-related equipment to detect biofouling. Changes to this program will be submitted to and approved by the NRC (both the Region and NRR) prior to implementation.

5.6 ~~6.9~~ REPORTING REQUIREMENTS

INSERT NEW PROGRAMS A4

INSERT NEW PROGRAMS M2

ROUTINE REPORTS

5.6 ~~6.9.1~~ In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Office of the NRC and a copy to the NRC Resident Inspector, unless otherwise noted.

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STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPE ONAL CONDITIONS or other conditions specified for individual Limiting Cor or Operation unless otherwise stated in an individual Surveillance

4.0.2 Each Surveillance Requirement shall be pe pecified surveillance interval with a maximum allowabl ceed 25 percent of the specified surveillance inte

4.0.3 Failure to perform a Surveill n allowed surveillance interval, defined by Specif constitute non-compliance with the OPERABILITY requiremen ndition for Operation. The time limits of the ACTION re able at the time it is identified that a SURVEILLANCE een performed. The ACTION requirements may be de urs to permit the completion of the surveillance when th me limits of the ACTION requirements are less than 24 requirements do not have to be performed on inoperable

Addressed in Chapter 3

4.0.4 E NAL CONDITION or other specified applicable condi unless the Surveillance Requirement(s) associated with ion for Operation have been performed within the applic. nce interval or as otherwise specified. This provision shall nc passage through or to OPERATIONAL CONDITIONS as required to comply wi ION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

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a. 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 shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program (ISI) for piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) shall be performed in accordance with the NRC positions included in Generic Letter 88-01.

LA9

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CONTAINMENT SYSTEMS 〈5.5.6〉

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent standby gas treatment subsy shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, ar

ACTION:

With one standby gas treatment subsy .ore the inoperable subsystem to OPERABLE status with . least HOT SHUTDOWN within the next 12 hours and ir .n the following 24 hours.

Addressed in Section 3.6

SURVEILLANCE REQUIREM

4.6.5.4 Ea ment subsystem shall be demonstrated OPERABLE:

a. r 31 days by initiating, from the control room, flow PA filters and charcoal adsorbers and verifying that em operates for at least 10 hours with the heaters

Or

LA1

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem, by:

5.5.6.a & 5.5.6.b ~~1.~~ Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05%, using the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and a system flow rate of 12,500 cfm ± 10%.

LA1

5.5.6.c ~~2.~~ Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

5.5.6 a 5.5.6 b & 5.5.6 d ~~3.~~ Verifying a subsystem flow rate of 12,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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(LAR 93-14R1)

 {5.5.6}

LA1

5.5.6 ~~c~~. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%. 5.5.6.c

d. At least once per 18 months by: LA1

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the: Addressed w/ Section 3.6

*a) LOCA, and

b) Annulus ventilation exhaust high radiation signal.

5.5.6.d ~~2~~. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while the filter train is operating at a flow rate of 12,500 cfm ± 10%.

Addressed w/ Section 3.6

3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

a) Manual initiation from the control room, and

*b) Simulated automatic initiation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5.5.6.e ~~5~~. Verifying that the heaters dissipate $\geq$ 61 kw when tested in accordance with ANSI N510-19~~80~~ 89 at the design supply voltage. L5 5.0 #142

5.5.6 ~~e~~. Verifying, after each complete or partial replacement of a HEPA filter bank, that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-19~~80~~ 89 while operating the system at a flow rate of 12,500 cfm ± 10%. LA1 5.5.6.a L5 5.0 #142

Addressed w/ Section 3.6

*The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

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<5.5.6>

5.5.6 ~~f.~~ Verifying, after each complete or partial replacement of a charcoal adsorber bank, that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-~~1980~~ for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 12,500 cfm ± 10%.

5.5.6.b

LA1

L5 — S.O #143

INSERT — A7 — The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

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<5.5.6>

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem, by: LA1

5.5.6.a ~~1.~~ & 5.5.6.b Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05%, using the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and a system flow rate of 10,000 cfm ± 10%.

5.5.6.c ~~2.~~ Verifying within 31 days after removal (LA1) that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl icdide penetration of less than 0.175%; and

5.5.6.a ~~3.~~ 5.5.6.b c 5.5.6.d Verifying a subsystem flow rate of 10,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-~~1980~~.9 LS 5.0 #143

5.5.6 ~~d.~~ After every 720 hours of charcoal adsorber operation, (LA1) by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Guide 1.52, Revision 2, March 1978, Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%. 5.5.6.c

e. At least once per 18 months by: LA1

Addressed w/ Section 3.6

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

**a) LOCA, and

b) Fuel Building ventilation exhaust high radiation signal.

5.5.6.d ~~2.~~ Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while the filter train is operating at a flow rate of 10,000 cfm ± 10%.

Addressed w/ Section 3.6

3. Verifying that the subsystem starts and isolation dampers actuate to isolate the normal flow path and to divert flow through the charcoal filters on each of the following test signals:

**The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

LAR 93-14R1

<5.5.6>

SURVEILLANCE REQUIREME. S (Continued)

Addressed w/ Section 3.6

a) Manual initiation from the control room, and

**b) Simulated automatic initiation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5.0 #143

LS

5.5.6.e ~~f.~~ Verifying that the heaters dissipate ≥49 kw when tested in accordance with ANSI N510-1980 9 at the design supply voltage.

LA1

5.5.6 ~~f.~~ Verifying, after each complete or partial replacement of a HEPA filter bank, that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 9 while operating the system at a flow rate of 10,000 cfm ± 10%.

5.5.6.a

LS 5.0 #143

LA1

5.5.6 ~~g.~~ Verifying, after each complete or partial replacement of a charcoal adsorber bank, that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 9 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 10,000 cfm ± 10%.

5.5.6.b

LS 5.0 #143

Addressed w/ Section 3.6

**The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

⟨5.5.6⟩

SURVEILLANCE REQUIREMENTS (Continued)

LA1 c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

5.5.6.a 5.5.6.b ~~1.~~ Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

5.5.6.c ~~2.~~ Verifying within 31 days after removal (LA1) that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and S.O #143

5.5.6.a 5.5.6.b 5.5.6.d ~~3.~~ Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-~~1980~~. L5 9

5.5.6 ~~d.~~ After every 720 hours of charcoal adsorber operation (LA1) by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

5.5.6.c

e. At least once per 18 months by: LA1

5.5.6.d ~~1.~~ Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

Addressed w/ Section 3.7

2. Verifying that on each of the below emergency mode actuation test signals, the subsystem automatically switches to the emergency mode of operation, the isolation valves close within 30 seconds, and the control room is maintained at a positive pressure of ≥ 1/8 inch water gauge relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to 4,000 cfm:

S.O #143

**a) LOCA, and

b) Local air intake radiation monitor - High.

L5

5.5.6.e ~~3.~~ Verifying that the heaters dissipate 23 ± 2.3 kw when tested in accordance with ANSI N510-~~1980~~, 9 at the design supply voltage.

Addressed w/ Section 3.7

**The specified 18 month interval during the first operating cycle may be extended to coincide with completion of the first refueling outage, scheduled to begin 9-15-87.

LAR 93-14R1

<5.5.6>

SURVEILLANCE REQUIREMENTS (Continued)

5.5.6.a ~~g~~. Verifying after each complete or partial replacement of a HEPA filter bank that the HEPA filter bank removes 99.95% of the DOP when they are tested in place in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%. LA1 9

5.5.6.b ~~g~~. Verifying after each complete or partial replacement of a charcoal adsorber bank that the charcoal adsorber bank removes 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in place in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%. LA1 9

L5

5.0
#143

LAR 43-14X1

PLANT SYSTEMS

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All required snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3, and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

a. With one or more required snubbers inoperable on a system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

b. Perform engineering evaluations per the applicable section of the approved ISI Program.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated to be OPERABLE by implementing the examination and test requirements of the approved ISI Program and as indicated in Specification 4.0.5. Only a previously approved revision of the ISI Program may be implemented. Subsequent revisions to the program shall be submitted to the NRC in accordance with the requirements of 10 CFR 50.55a(g).

LA1

9

5.0
#36

⟨5.5.8⟩

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying the diesel generator is aligned t e standby power to the associated emergency busse

7. Verifying the pressure in all d t receivers to be greater than

b. At least once per 24 hour generator 1A and 1B that the lube oil ng and that the lube oil sump heater an are OPERABLE.

Addressed in Section 3.8

c. By removin

1. st once per 31 days and after each esel is operated for greater than 1 hour,

2. Fr storage tank at least once per 31 days.

LA2

d. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1. By verifying in accordance with the tests specified in ASTM D975-81, prior to addition to the storage tanks, that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2. By verifying an antioxidant type diesel fuel oil stabilizer is added to new fuel added to the storage tanks in accordance with manufacturer's recommendations.

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<5.5.8>

SURVEILLANCE REQUIREMENTS (Continued)

SDC

LA2

3. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81, except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78 and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

f. At least once per 18 months#****, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance w' procedures prepared in conjunction with its manufact recommendations for this class of standby service

2. Verifying the diesel generator capability to greater than or equal to 917.5 kw for dies' greater than or equal to 509.2 kw for d' and greater than or equal to 1995 kw for , while maintaining engine speed less than . the difference between nominal speed an' setpoint or 15% above nominal, whicheve'

3. Verifying the diesel gene' reject a load of 3030-3130 kw*** for di' and 1B and 2500-2600 kw*** for diesel generator g. The generator voltage shall not exceed 4' l generator 1A and 1B and 5400 volts for ' during and following the load rejection.

4. Simulati' e power by itself, and:

a) II:

g deenergization of the emergency busses and nedding from the emergency busses.

Addressed in Section 3.8

#' diesel, the diesel must be operated with a load in the manufacturer's recommendations.

ansients due to changing bus loads shall not invalidate the

4.8.1.1.2.f.1 to be performed every refueling outage, for diesel rators 1A and 1B only.

<5.5.5>

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figu .1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and s ed with:

a. A k_{eff} less than or equal to 0.95, when flo ted water, including all calculational uncertainties described in Section 9.1 of the FSAR.

b. A fuel assembly center-to-center sto 7 in. within rows and 12.25 in. between rows in the age Racks in the upper containment pool.

c. A fuel assembly center-to-cent ng of 6.28 in., with a neutron poison material betw es, in the High Density Storage Racks in the spent cility in the Fuel Building.

The storage of spent fuel in inment fuel storage pool is prohibited during OPERATION and 2.

Addressed in Chapter 4.0

5.6.1.2 For the first core for new fuel stored dry in the spent fuel storage racks sh ratively controlled to not exceed 0.98 when optimum moderat , fogging, or small droplets) is assumed.

5.6.1.3 Provisions o avoid the entry of sources of optimum moderation (foam, or small droplets) to preclude that K_{eff} for new fuel, stored storage facility, could exceed 0.98.

DRAINAGE

5.6.2 The age pool is designed and shall be maintained to prevent inadverte he pool below elevation 95'.

CAPACI

5.6 el storage pool in the fuel building is designed and shall be a storage capacity limited to no more than 2680 fuel assem y fuel manufactured by General Electric may be stored in the spent l.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.5.5

~~5.7.1~~ The components identified ~~in Table 5.7.1-1~~ are ~~designed and shall be~~ maintained within the cyclic or transient limits ~~of Table 5.7.1-1.~~

LA2

LA2

RIVER BEND - UNIT 1

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	120 heatup and cooldown cycles	70°F to 560°F to 70°F
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	40 hydrostatic pressure or leak tests	Pressurized to ≥ 930 psig and ≤ 1250 psig

LA2

INSERT
~~5-7~~
6-15 (14)

<5.5>

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STARTUP REPORT (Continued)

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 unit.

LA3

6.9.1.2 The startup report shall address each of the tests identified in Final Safety Analysis Report (Section 14.2.12.2) 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 required in license conditions based on other commitments shall be included in this report.

6.9.1.3 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 operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

5.6.1 ~~6.9.1.4~~ Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March ~~1~~ 31 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality. (A4) (A9)

5.6.1 ~~6.9.1.5~~ Reports required on an annual basis shall include:

~~a.~~ A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[a] (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent 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 should be assigned to specific major work functions;

5.6.4 ~~b.~~ Documentation of all challenges to safety/relief valves and a summary of SRV failures.

A12

5.6.1 [a]This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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ANNUAL REPORTS (Continued)

A2

C. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (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 limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than 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 isotope 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.

MONTHLY OPERATING REPORTS

5.6.4 Routine reports of operating statistics and shutdown experience, including documentation of all failures to the main steam system safety/relief valves, shall be submitted on a monthly basis no later than the ~~15th~~ 1st of each month following the calendar month covered by the report. A4

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

5.6.2 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality. A9

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate) with preoperational studies, operational controls and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. LA1

The Annual Radiological Environmental Operating Reports shall include the results of all analyses of radiological environmental samples and of all environmental radiation measurements taken during the report period pertinent to the locations specified in the tables and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining

LAR 14R1

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

LA1

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 4.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

ANNUAL EFFLUENT RELEASE REPORT

5.6.3 ~~6.9.1.8~~ Routine Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted ~~within 90 days after January 1~~ by May 1 of each year. A4

The Annual Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

LA1

The Annual Radioactive Effluent Release Report shall include a summary of hourly meteorological data collected over the previous year. This summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction and atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. The report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.3) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters of the OFFSITE DOSE CALCULATION MANUAL (ODCM).

LA1

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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ANNUAL EFFLUENT RELEASE REPORT (Continued)

The Annual Radioactive Effluent Release Report shall also include an assessment of radiation doses to the likely most-exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," Rev. 0, October 1978.

LA1 The Annual Radioactive Effluent Release Report shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Annual Radioactive Effluent Release Report shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the ODCM, as well as a listing of new locations for dose calculations and environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

5.0 #55

A10

SPECIAL REPORTS

6.9.2 Special reports shall be submitted in the following manner:

a. Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Office of the NRC and a copy to the NRC Resident Inspector, within the time period specified for each report.

A14

b. Special reports in regard to Corbicula will be submitted to the NRC within 30 days of identification of infestation. In accordance with the settlement agreement dated October 10, 1984, these reports shall describe the level of infestation, affected systems and measures taken to prevent further infestation.

CORE OPERATING LIMITS REPORT

INSERT SPECIAL REPORT 1

5.6.5.a ~~6.9.3.1~~ Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

A13

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specification 3.2.1.

LAR 93-14R1

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

ACTION 80 - a. With the number of OPERABLE accid itoring instrumentation channels less than the Requ r of Channels shown in Table 3.3.7.5-1, restore th hannel(s) to OPERABLE status within 7 day ast HOT SHUTDOWN within the next 12 hours a WN within the following 24 hours.

b. With the number of O monitoring instrumentation channels less . Channels OPERABLE requirements of Table 3 the inoperable channel(s) to OPERABLE st urs or be in at least HOT SHUTDOWN wit ours and in COLD SHUTDOWN within the follow

Addressed in Section 3.3

ACTION 81 - With the num hannels less than required by the Minimum Ch requirement, either restore the inoperable Cha LE status within 72 hours, or:

a. Initia planned alternate method of monitoring the appropr arameter(s), and

b. Prepare and submit, within 14 days following the event, a Special Report to the Commission, pursuant to Specification 6.9.2, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

LA3

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EMERGENCY CORE COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For ECCS divisions I and II, provided that ECCS ' III is OPERABLE:

1. With LPCI subsystem "A" and either LPC' or "C" inoperable, restore at least the inor system "A" or inoperable LPCI subsystem "B" or status within 72 hours.

2. With the LPCS system inoperabl I subsystem "B" or "C" inoperable, restore a' erable LPCS system or inoperable LPCI subsyst OPERABLE status within 72 hours.

3. Otherwise, be in at l within the next 12 hours and in COLD SHUTDOWN owing 24 hours*.

e. For ECCS divisions I that ECCS division III is OPERABLE and divisi otherwise OPERABLE:

1. With one of red ADS valves inoperable, restore the inoper to OPERABLE status within 14 days or be in at OWN within the next 12 hours and reduce reacto ssure to ≤ 100 psig within the next 24 h

2. W of the above required ADS valves inoperable, HOT SHUTDOWN within 12 hours and reduce reactor essure to ≤ 100 psig within the next 24 hours.

Addressed in Section 3.5

f. W scharge line "keep filled" pressure alarm inst on channel inoperable, perform Surveillance Requirement 4.5.1.a. least once per 24 hours.

LA3

g. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted within 90 days to the Commission, pursuant to Specification 6.9.2, 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.

Addressed w/ Section 3.5

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

2. At least once per 12 hours with eported room temperature <50°F.

h. At least once per 10 years, or ations which could affect diesel generator int arting all three diesel generators simultaneously , and verifying that all three diesel generator least 450 rpm for diesel generators 1A and 1B iesel generator 1C in less than or equal to 1

Addressed in Section 3.8

i. At least once

1. Drai storage tank, removing the accumulated se ng the tank using a sodium hypochlorite or on, and

2. Pe pressure test of those portions of the diesel fuel oil designed to Section III, subsection ND of the ASME Code, accordance with ASME Code Section XI Article IWD-5000.

LA3

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission, pursuant to Specification 6.9.2, within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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CORE OPERATING LIMITS REPORT (Continued)

A13

b. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

c. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.4.

d. The REACTOR PROTECTION SYSTEM (RPS) response time for APRM thermal time constant for Specification 3.3.1.

and shall be documented in the CORE OPERATING LIMITS REPORT (COLR).

5.6.5.b ~~6.9.3.2~~ The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (the approved version at the time the reload analyses are performed).* A15

5.6.5.c ~~6.9.3.3~~ The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits and nuclear limits such as shutdown margins, and transient and accident analysis limits) of the safety analysis are met.

5.6.5.d ~~6.9.3.4~~ The CORE OPERATING LIMITS REPORT, including any mid-cycle revision or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

INSERT 5.6.6 PTLR (per NUREG Markup) M2

LA1

6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 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.

A15

~~*For Cycle 5 only, the result of misoriented fuel bundle analysis need not be considered in determining the OLMCPR due to the extensive verifications performed of fuel bundle placement and orientation for Reload 4.~~

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RECORD RETENTION (Continued)

LA1

c. All REPORTABLE EVENTS

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

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 emergency drills and exercises.

6.10.3 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 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual that are not listed in Specification 6.10.1.

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RECORD RETENTION (Continued)

LA1

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 FRC and the NRB.

l. Records of the service lives of all snubbers, including the date at which the service life commences, and associated installation and maintenance records.

m. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at the specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

5.7 ~~6.12~~ HIGH RADIATION AREA

5.7.1 ~~6.12.1~~ In lieu of the "control device" or "alarm signal" required by paragraph 20.~~203(c)(2)~~ 1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. 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.

c. An individual qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and who shall perform periodic radiation surveillance at the frequency specified in the RWP by the Health Physicist.

*Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

LAR 93-14R1

HIGH RADIATION AREA (Continued)

5.7.2 ~~6.12.2~~ In addition to the requirements of Specification 6.12.1, accessible areas with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Control Operating Foreman on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of access under an approved RWP that specifies the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For accessible areas that are located within large areas, such as the containment, where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, and within which radiation levels are such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem,* then that area shall be

5.7.3 roped off and conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

1. Shall be submitted to the Commission in the Annual Radioactive Effluent Release Report. This submittal shall contain:

LA1

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Specification 6.5.2.

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2.

INSERT CTS 1.34

*Measurement made at ≤ 18 inches from source of radioactivity. LA2

LAR 93-14R1

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a 'e fault in a reactor coolant system component body, pipe wall or

PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING

1.32 PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING sh :

a. All containment penetrations required to ng accident conditions are closed by at least one m ind flange, or deactivated automatic valve secured position. Up to twelve vent and drain line path ened under administrative control for the purp lance testing provided the total calculated flow the open vent and drain line pathways is less than .2 cfm.

b. All containment hatches are cl

c. Each containment air lock is with the requirements of Specification 3.6.1.4.

PRIMARY CONTAINMENT INTEGRITY - OPE

1.33 PRIMARY CONTAINMENT INTEGRIT shall exist when:

a. All containment pen red to be closed during accident conditions are eit

1. Capable of an OPERABLE containment automatic isolation

2. Closed b manual valve, blind flange, or deactivated secured in its closed position, except as pr fication 3.6.4.

b. All cont nt hatches are closed and sealed.

c. Each c lock is in compliance with the requirements of Spec .4.

d. Th eakage rates are within the limits of Specification

e. ion pool is in compliance with the requirements of Speci- 6.3.1.

f. ing mechanism associated with each primary containment penetratio e.g., welds, bellows or O-rings, is OPERABLE.

Addressed in Chapter 1.0

PROCESS CONTROL PROGRAM (PCP)

1.34 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71 and

LA1

LAR 93-14R1

1A1 Federal and State regulations and other requirements governing the disposal of the radioactive waste. DOC

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transf the reactor coolant of 2894 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the om when the monitored parameter exceeds its trip setpoint at sor until de-energization of the scram pilot valve solenoids time may be measured by any series of sequential, overlappir s such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those ified in 10 CFR 50.73.

ROD DENSITY

1.38 ROD DENSITY shall be the number of ches inserted as a fraction of the total number of contro All rods fully inserted is equivalent to 100% ROD DENSITY.

Addressed in Chapter 1.0

SECONDARY CONTAINMENT INTEGRITY -

1.39 SECONDARY CONTAINMENT INT ILDING shall exist when:

a. All Fuel Buildinc equired to be closed during accident conditions valves, blind flanges, or dampers secured in position.

b. All Fuel B t hatch covers are installed.

c. The Fue coal Filtration System is in compliance with the r pecification 3/4.6.5.6.

d. At in each access to the Fuel Building is closed, ne entry and exit of personnel and equipment.

e. within the Fuel Building is maintained in complianc quirements of Specification 4.6.5.1.a.

SE ENT INTEGRITY - OPERATING

CONTAINMENT INTEGRITY - OPERATING shall exist when:

ll Auxiliary Building penetrations, Fuel Building penetrations and Shield Building annulus penetrations required to be closed during accident conditions are either:

LAR 93-14.1

5.5.1

~~6.14~~ OFFSITE DOSE CALCULATION MANUAL (ODCM)

INSERT CTS 1.27

~~6.14.1~~ The ODCM shall be approved by the Commission prior to implementation.

~~6.14.2~~ Licensee-initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Annual Radioactive Effluent Release Report. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed, with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   A6

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Specification 6.5.2. LA1

   5.0 #64

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2. LA1

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report and approved pursuant to Specification 6.5.2. The discussion of each change shall contain:

   A2

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

   b. Sufficiently detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

(LAR 93-14R1)

of a logic circuit, from sensor through and incl[illegible] he actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTION[illegible] be performed by any series of sequential, overlapping or total [illegible]h that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBL[illegible] [illegible]ersons who are not occupationally associated with [illegible] [illegible]gory does not include employees of the utility, its co[illegible] Also excluded from this category are persons who ent[illegible] [illegible] equipment or to make deliveries. This category do[illegible] [illegible] use portions of the site for recreational, occupational [illegible] associated with the plant.

Addressed in Chapter 1.0

5.0 #69

MINIMUM CRIT[illegible]

1.26 The MINIM[illegible] [illegible]CAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the cor[illegible]

5.5.1

LA1

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.27 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. It shall also contain a table and figure defining current radiological environmental monitoring sample locations.

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or d[illegible]ice shall be OPERABLE or have OPERABILITY when it is capable of perfor[illegible] [illegible]s specified function(s) and when all necessary attendant instrumen[illegible] [illegible]trols, electrical power, cooling or seal water, lubrication or o[illegible] [illegible]quipment that are required for the system, subsystem, [illegible] [illegible]device to perform its function(s) are also capable of pe[illegible] [illegible]ed support function(s).

OPERATIONAL CONDITION - COND[illegible]

1.29 An OPERATIONAL CO[illegible] [illegible]ION, shall be any one inclusive combination of mode [illegible] average reactor coolant temperature as specified in Tabl[illegible]

Addressed in Chapter 1.0

PHYSICS TEST[illegible]

1.30 PHYS[illegible] [illegible]e those tests performed to measure the fundamental nuclear cha[illegible] of the reactor core and related instrumentation and 1) described in [illegible]14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or [illegible] otherwise approved by the Commission.

d. An evaluation of the change, which shows (1) the predicted releases of radioactive materials in liquid and gaseous effluents and/or (2)-quantity of solid waste, that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures, to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population, that differ from those previously estimated in the license application and amendments thereto;

f. 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;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

A2

h. Documentation of the fact that the change was reviewed and found acceptable pursuant to Specification 6.5.2.

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2.

INSERT CTS
3/4.11 and 3/4.12

A3

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

<RELOCATED> R1

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1.1.1-1

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

LAR 93-14R1

TABLE 4.11.1.1.1-1

R1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release (Liquid Radwaste Recovery Sample Tanks[b])	P Each Batch	P Each Batch	Principal Gamma Emitters[c]; except for Ce-144	5×10^{-7} 5×10^{-6}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

TABLE 4.11.1.1.1-1 (continued)

R1

TABLE NOTATION (continued)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

 is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

b - A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, ZN-65, Mo-99, Cs-134, Cs-137, CE-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

d - A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

DOSE

<RELOCATED>

R2

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose, from the release of radioactive materials in liquid effluents, exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined, at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

<RELOCATED>

R3

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses, due to the liquid effluent, to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

⟨5.5.7⟩

5.0
#109

LIMITING CONDITION FOR OPERATION

5.5.7
~~3.11.1.4~~ The quantity of radioactive material contained in any unprotected outdoor tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

LA1

ACTION:

a. With the quantity of radioactive material in any of the above unprotected outdoor tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank; within 48 hours reduce the tank contents to within the limit; and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

5.5.7
~~4.11.1.4~~ The quantity of radioactive material contained in each of the above unprotected outdoor tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank. LA1

A4 INSERT

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the explosive gas and storage tank radioactivity monitoring program surveillance frequencies.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

<RELOCATED> (R4)

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate, due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate, due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents, shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2.1.2-1.

INSERT
3/4 11-8
6-24(8)

TABLE 4.11.2.1.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Main Plant Exhaust Duct	M[c] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
B. Fuel Building Ventilation Exhaust Duct	M[d] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Radwaste Building Ventilation Exhaust Duct	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
D. All Release Types as listed in A, B, C above.	Continuous[e]	W[f] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[e]	W[f] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	SR-89, SR-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

R4

LAR 93.14R1

)

TABLE 4.11.2.1.2-1 (Continued)

R4

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size, in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

TABLE 4.11.2.1.2-1 (Continued)

R4

TABLE NOTATION

b - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

c - Sampling and analysis shall also be performed, within one hour following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER, unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the condenser offgas noble gas activity monitor shows that offgas activity has not increased by more than a factor of 3.

d - Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

e - The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

f - Samples shall be changed at least once per 7 days, and analyses shall be completed within 48 hours after changing or after removal from sampler. Sampling shall also be performed at least once per 24 hours for the main plant exhaust for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the condenser offgas noble gas monitor shows that offgas activity has not increased more than a factor of 3.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

<RELOCATED> R5

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 At least once per 31 days, cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM.

LAR 93-14R1

<RELOCATED> R6

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose 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 to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release, in gaseous effluents, of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 At least once per 31 days, cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, shall be determined in accordance with the methodology and parameters in the ODCM.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

<RELOCATED> R7

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM inoperable for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

 1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

 2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

 3. Summary description of action(s) taken to prevent recurrence.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 The instruments specified in the ODCM shall be checked every 12 hours, whenever the main condenser air ejector is in operation, to ensure that the GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is functioning.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT <RELOCATED> R8

LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses, due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), would exceed 0.3 mrem to any organ in a 31 day period.

APPLICABILITY: At all times other than when the VENTILATION EXHAUST TREATMENT system is undergoing routine maintenance.

ACTION:

a. With gaseous waste being discharged from the ventilation exhaust ducts without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 Doses due to gaseous releases from the site shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

<5.5.7>

LIMITING CONDITION FOR OPERATION

5.5.7 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume. LA1

APPLICABILITY: Whenever the main condenser offgas treatment system is in operation. LA1

ACTION: LA1

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

5.5.7 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring, with the hydrogen monitor(s) required in compliance with Specification 3.3.7.11, the waste gases in the main condenser offgas treatment system whenever the main condenser evacuation system is in operation. LA1

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The release rate of the sum of the activities from th s* measured prior to the holdup pipe shall be limited to less th o 290 millicuries/sec after 30 minutes decay.

APPLICABILITY: Whenever the main condenser offgas treatr in operation.

ACTION:

With the release rate of the sum of the activitier le gases* prior to the holdup pipe exceeding 290 millicuries/sec utes decay, restore release rate to within its limit within 72 hour least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

Addressed in Section 3.7

4.11.2.7.1. The radioactivity of nob r to the holdup pipe shall be continuously monitored in accord ecification 3.3.7.1.

4.11.2.7.2 The release rate of e activities from the noble gases* measured prior to the h all be determined, at the following frequencies, to be within the ecification 3.11.2.7 by performing an isotopic analysis of a r sample of gases taken prior to the holdup pipe.

a. At least once

b. Within 4 h g an increase, as indicated by the Noble Gas Activity greater than 50%, after factoring out increases due to RMAL POWER level, in the nominal steady state fissio from the primary coolant.

c. The f Specification 4.0.4 are not applicable.

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*Gamma on detectors are used to measure the Kr-85m, -87, -88 and Xe-133, 135, -138 contribution after 30 minutes decay.

LAR 93-14R1

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE <RELOCATED> R9

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

LAR 93-14R1

3/4.11.4 TOTAL DOSE

<RELOCATED> R10

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and to radiation from uranium fuel cycle sources, shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made, including direct radiation contributions from the reactor units and from outside storage tanks, to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and that includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits and, if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3 and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor unit and from any unprotected outdoor storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, Action a.

LAR 93-14R1

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

<RELOCATED> R11

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity, as the result of plant effluents, in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or broad leaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The

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*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LAR 93-14R1

R11

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Annual Radioactive Effluent Release Report and include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12.1-1 and the detection capabilities required by Table 4.12.1-1.

INSERT ~~3/4 12-9~~ 6-24(21)

TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: 3 samples from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average groundlevel D/Q.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following

R11

LAR 93-14R1

INSERT
3/4 12-4
6-24(22)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
	1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q.		filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
	1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]		
3. WATERBORNE			
a. Surface[f]	1 sample upstream and 1 sample downstream.	Weekly samples composited monthly or quarterly.	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
	Discharge line.	Composite sample over 1-month period[g]	
b. Ground	Samples from 1 or 2 sources only if likely to be affected[h]	Quarterly	Gamma isotopic[e] and tritium analysis quarterly.
c. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture[illegible] monthly at other ti[illegible]

R11

LAR 93-14R1

INSERT
~~3/4 12-5~~
6-24(23)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
a. Milk (cont'd)	none, then 1 sample from milking animals in each of 3 areas 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.[i] 1 sample from milking animals at a control location 15-30 km distant and in the least prevalent wind direction.		
b. Fish and Invertebrates	1 sample of each of three commercially and/or recreationally important species in vicinity of plant discharge area. 1 sample of each of three species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis[e] on edible portions.
c. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near each of two different locations near the site boundary of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly during the growing season.	Gamma isotopic[e] and I-131 analysis.

LAR 93-14R1

R11

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Continued	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant near the least prevalent wind direction, if milk sampling is not performed.	Monthly during the growing season.	Gamma isotopic[e] and I-131 analysis.

INSERT
~~3/4 12-6~~
6-24 (24)

R11

LAR 93-14R1

INSERT ~~3/4 12-7~~ 6-24(25)

TABLE 3.12.1-1 (Continued)

TABLE NOTATION

a - The ODCM shall include, in a table and figures, specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, for each sample location in Table 3.12.1-1. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples, and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b - One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

c - The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

d - Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

R11

LAR 93-14R1

INSERT ~~3/4 12-8~~ 6-24 (26)

TABLE 3.12.1-1 (Continued)

TABLE NOTATION (Continued)

e - Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

f - The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence.

g - Composite samples shall be collected at intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

h - Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

i - The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

R11

LAR 93-14R1

INSERT ~~3/4 12-9~~ 6-24(27)

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water[(a)] (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Nb-95	400				
Zr-95	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			300	
La-140	200			300	

(a)For discharge line samples, these values may be increased by a factor of 11.4 to account for near-field dilution by the Mississippi River.

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

**If no drinking water pathway exists, a value of 20 pCi/ℓ may be used.

R11

LAR 93-14R1

RIVER BEND - UNIT 1

INSERT
~~3/4 12-10~~
6-24 (2B)

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Nb-95	15					
Zr-95	30					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
La-140	15			15		
Ba-140	60			60		

*If no drinking water pathway exists, a value of 3000 pCi/ℓ may be used.

**If no drinking water pathway exists, a value of 15 pCi/ℓ may be used.

LAR 93-14R1

R11

LAR 93-14R1

TABLE 4.12.1-1 (Continued)

R11

TABLE NOTATION

a - This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

b - Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size, in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

LAR 93-14R1

R11

TABLE 4.12.1-1 (Continued)

TABLE NOTATION (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a partic r measurement. Analyses shall be performed in such a manner that t.. stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

LAR 93-14R1

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

<RELOCATED> R12

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location, in each of the 16 meteorological sectors, of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

b. With land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Annual Radioactive Effluent Release Report and include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 At least once per 12 months, the land use census shall be conducted during the growing season, using that information that will provide the best results, such as by a door-to-door survey or aerial survey or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*In lieu of the garden census, broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs. Specifications for broad leaf vegetation sampling in Table 3.12.1-1, 4c shall be followed, including analysis of control samples.

LAR 93-14R1

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

<RELOCATED> R13

3.12.3 Analyses shall be performed on radioactive materials, that correspond to samples required by Table 3.12.1-1, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report to the Commission, in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7, the corrective actions taken to prevent a recurrence.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

LA? 93-14R1

ATTACHMENT 1B

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14RT

DISCUSSION OF CHANGES
CTS: 6.1 - RESPONSIBILITY

ADMINISTRATIVE

A.1 Where possible, plant specific management position titles in the current Technical Specifications are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed Specification 5.3, and compliance details relating to the plant specific management position titles are identified in licensee controlled documents (such as the USAR). The two major specific replacements are the generic "plant manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the corporate executive responsible for overall plant nuclear safety" in place of the Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (such as the USAR). This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative.

5.0 #2

A.2 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

5.0 #2

LA.1 The requirement for the Senior Vice President to issue a directive on an annual basis identifying the Shift Supervisor as having responsibility for control room command function has been relocated to plant controlled documents. Control of this requirement under 10 CFR 50.59 controls is adequate.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.2 - ORGANIZATION

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This statement only provides an example that constitutes the accepted shift period. The Technical Specification continues to require specific guidelines on overtime requirements. Since this is only an example and does not change the actual requirements, this change is considered administrative.

A.3 This comment number is not used for this station.

A.4 Where possible, plant specific management position titles in the current Technical Specifications are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed Specification 5.3, and compliance details relating to the plant specific management position titles are identified in licensee controlled documents (such as the USAR). The two major specific replacements are the generic "plant manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the corporate executive responsible for overall plant nuclear safety" in place of the Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (such as the USAR). This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative.

A.5 The details of the methods for implementing this requirement are contained in the Commission's Policy Statement. Repeating these details is redundant and unnecessary, and the omitted information provides no additional restriction or requirement. Therefore, this requirement is not repeated in the proposed Technical Specifications. Since the requirements remain the same, this change only affects the presentation method and is considered administrative.

DISCUSSION OF CHANGES
CTS: 6.2 - ORGANIZATION

ADMINISTRATIVE
(continued)

A.6 These specific fire protection issues are covered by a more generic program (i.e., Fire Protection Program) which provides for the definition and implementation of these details. Therefore, it is not necessary to specifically identify each of these fire protection issues in the Technical Specifications. Consequently, removing these issues from the Technical Specifications is considered an administrative change.

A.7 This comment number is not used for this station.

A.8 This comment number is not used for this station.

A.9 This comment number is not used for this station.

A.10 The requirement for an SRO to be present during fuel handling and core alterations is contained in 10 CFR 50.54. Therefore, there is no need to repeat these requirements in the Technical Specifications. Since the requirements remain the same, this change only affects the presentation method and is considered administrative.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

M.2 This comment number is not used for this station.

M.3 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.2 - ORGANIZATION

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 Details of the Operator License requirements for these specific positions are relocated to the USAR and procedures although a requirement that the Operations Supervisor have an SRO License is retained in Section 5.2.2.f. Additionally, changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.2 This comment number is not used for this station.

LA.3 The staffing requirements of Table 6.2.2-1 are relocated to the USAR and procedures. The requirements of Table 6.2.2-1 are removed for the Technical Specifications and will be controlled by the licensee's administrative controls. 10 CFR 50.54 provides the requirements for shift complement regarding licensed operators. Additionally, the Technical Specifications will continue to specify when a licensed operator must be in the control room. The Table 6.2.2-1 requirements associated with the auxiliary operators are retained as 5.2.2.a with the associated allowance for unexpected absences retained in 5.2.2.c. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.4 This comment number is not used for this station.

LA.5 The requirements relating to the Independent Safety Engineering Group (ISEG) are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.6 This comment number is not used for this station.

"Specific"

L.1 This comment number is not used for this station.

L.2 This comment number is not used for this station.

5.0 #4

LA.7 Details of the site fire brigade requirements are moved to the USAR and procedures.

LA.8 This comment number is not used for this station.

LA.9 The definition of "normal work week" has been relocated from the Specifications and will be controlled under 10 CFR 50.59 controls

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.3 - UNIT STAFF QUALIFICATIONS

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 Where possible, plant specific management position titles in the current Technical Specifications are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed Specification 5.3, and compliance details relating to the plant specific management position titles are identified in licensee controlled documents (such as the USAR). The two major specific replacements are the generic "plant manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the corporate executive responsible for overall plant nuclear safety" in place of the Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (such as the USAR). This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

DISCUSSION OF CHANGES
CTS: 6.3 - UNIT STAFF QUALIFICATIONS

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 This comment number is not used for this station.

LA.2 This comment number is not used for this station.

LA.3 The requirements relating to reactor operator training and licensing program are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

L.1 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.4 - TRAINING

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This comment number is not used for this station.

A.3 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The requirements relating to the retraining and replacement program are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.5 - REVIEW AND AUDIT

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This comment number is not used for this station.

A.3 This comment number is not used for this station.

A.4 This comment number is not used for this station.

A.5 This comment number is not used for this station.

A.6 This comment number is not used for this station.

A.7 This comment number is not used for this station.

A.8 This comment number is not used for this station.

A.9 This comment number is not used for this station.

A.10 This comment number is not used for this station.

A.11 This comment number is not used for this station.

A.12 This comment number is not used for this station.

A.13 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGE - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

M.2 This comment number is not used for this station.

M.3 This comment number is not used for this station.

M.4 This comment number is not used for this station.

M.5 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.5 - REVIEW AND AUDIT

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA.1 This comment number is not used for this station.

LA.2 This comment number is not used for this station.

LA.3 This comment number is not used for this station.

LA.4 Review and Audit requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

L.1 This comment number is not used for this station.

L.2 This comment number is not used for this station.

L.3 This comment number is not used for this station.

L.4 This comment number is not used for this station.

(LAR 93-14R1)

DISCUSSION OF CHANGES
CTS: 6.6 - REPORTABLE EVENT ACTION

ADMINISTRATIVE

A.1 This requirement is contained in Title 10 of the Code of Federal Regulations. Repeating the requirements of the regulations is redundant and unnecessary, and creates an unnecessary burden to revise the Technical Specifications when the regulations change. Therefore, this requirement is not repeated in the proposed Technical Specifications. Since the requirements remain the same, this change only affects the presentation method and is considered administrative.

A.2 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Review and Audit requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

L.1 This comment number is not used for this station.

(LAR 93-1421)

DISCUSSION OF CHANGES
CTS: 6.7 - SAFETY LIMIT VIOLATION

ADMINISTRATIVE

A.1 The technical content of this requirement is being moved to another chapter of the proposed Technical Specifications in accordance with the format of the BWR Standard Technical Specification, NUREG-1434. Any technical changes to this requirement will be addressed with the content of the proposed chapter location.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Review and Audit requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.8 - PROCEDURES AND PROGRAMS

ADMINISTRATIVE

A.1 These types of procedures are required by the item immediately preceding which references Regulatory Guide 1.33. Therefore, it is not necessary to specifically identify each type of procedure. Since the requirements remain, this is considered to be a change in the method of presentation only, and therefore, is considered an administrative change.

A.2 Procedures to implement the Emergency Plan and the Security Plan are required by 10 CFR 50, Appendix E and 10 CFR 50.54(p). Since conformance with 10 CFR Chapter I is a license condition and the Emergency Plan and Security Plan are required to be implemented by 10 CFR Chapter I, specific identification of these plans is unnecessary duplication. This is a change in the presentation of the requirements only, and therefore, is considered an administrative change.

A.3 These specific programs are covered by a more generic item which requires this activity for all Programs and Manuals. Therefore, it is not necessary to specifically identify each program. Since the requirements remain, this is considered to be a change in the method of presentation only, and therefore, is considered an administrative change.

A.4 The technical content of several requirements are being moved fr another chapter of the current Technical Specifications and are proposed to be identified as Programs in accordance with the format of the BWR Standard Technical Specification, NUREG-1434. Other Programs currently identified in the Administrative Controls section are consolidated into this section. Any technical changes to the requirements are identified in their respective markups and addressed as indicated. These Programs include:

PSTS	CTS	
5.5.1	6.14 & 1.27	Offsite Dose Calculation Manual
5.5.2	6.8.4.a	Primary Coolant Sources Outside Cont.
5.5.3	6.8.4.c	Post Accident Sampling
5.5.5	5.7.1	Component Cyclic or Transient Limit
5.5.6	4.6.5.4, 4.6.5.5, & 4.7.2	Ventilation Filter Testing Program
5.5.7 8	3/4.11.1.4 & 3/4.11.2.6	Explosive Gas and Storage Tank Radioactive Monitoring Program
5.5.9	4.8.1.1.2.c	Diesel Fuel Oil Testing Program

A.5 This comment number is not used for this station.

A.6 This comment number is not used for this station.

A.7 An statement of applicability of SR 3.0.2 or SR 3.0.3 is needed

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.8 - PROCEDURES AND PROGRAMS

ADMINISTRATIVE
(continued)

to maintain the current allowances for surveillance frequency extensions since these SRs are not normally applied to frequencies identified in the Administrative Controls section of the Technical Specifications. Since this change maintains current requirements, it is considered a change of presentation method only.

A.8 This comment number is not used for this station.

5.0 #28

A.9 ~~The deletion of the Biofouling Prevention and Detection Program is being proposed based upon a River Bend Station submittal dated January 14, 1994 (RBG-39896). The program is no longer necessary based upon improvements to the service water system.~~

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

5.0 #116

M.2 ~~Four~~ Three new programs are included in the proposed Technical Specifications. These programs include:

5.5.4 Radioactive Effluent Controls Program
~~5.5.9 Fire Protection Program~~
5.5.~~10~~ 11 Technical Specification Bases Control
5.5.~~11~~ 10 Safety Function Determination Program

The Radioactive Effluent Controls Program has been included as a result of the revisions due to (1) the relocation of the radiological Technical Specifications consistent with Generic Letter 89-01 and (2) the changes to 10 CFR 20. ~~The Fire Protection Program is included to specifically identify the controls that support implementation of the requirements of 10 CFR 50.48. The Fire Protection Program has been included as a result of the revisions due to the relocation of the fire protection technical specifications consistent with Generic Letters 86-10 and 88-12.~~ The Bases Control program is provide to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program is included to support implementation of the support system operability characteristics of the Technical Specifications.

M.3 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.8 - PROCEDURES AND PROGRAMS

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Details of the methods for implementing this specification are relocated to the USAR and procedures. The guidance documents which dictate the methods are also identified in the USAR. Additionally, changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.2 Details of the methods for implementing this specification are relocated to the USAR, procedures and the proposed Administrative Controls section of the Technical Specifications as a Program. The guidance documents which dictate the methods are also identified in the USAR. Additionally, changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59, and the Program requirements are controlled as a proposed Technical Specification.

LA.3 This comment number is not used for this station.

LA.4 Review and ~~Audit~~ approval requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.5 This comment number is not used for this station.

LA.6 Requirements relating to the Process Control Program are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.7 Requirements relating to the In-Plant Radiation Monitoring Program are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.8 This comment number is not used for this station.

LA.9 Requirements relating to the inservice inspection and testing are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

(LAR 93-14R1)

DISCUSSION OF CHANGES
CTS: 6.8 - PROCEDURES AND PROGRAMS

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

"Specific"

L.1 This comment number is not used for this station.

5.0 #30

L.2 The intent of NUREG-0737 was to address "emergency operating procedures", consequently the existing TS wording was revised appropriately to limit the scope of procedures requiring consideration of NUREG-0737.

L.3 This comment number is not used for this station.

L.4 The proposed change deletes the program associated with the prevention and detection of Asiatic Clams (Corbicula) based upon improvements to the non-safety related Normal Service Water System (SWS). The proposed change is acceptable based on the following:

5.0 #28

1. The source of makeup water to the SWS is no longer the Mississippi River, which is the source of Asiatic Clams. Demineralized water or well water is used.

2. The possibility of the SWS becoming contaminated by any other means is highly unlikely since it is a "closed-loop" system.

3. Contamination of the Standby Cooling Tower basin through avian transport is highly unlikely, Moreover, the Standby Cooling Tower basin water is not conducive to Asiatic Clam survival.

4. Post RF-4 inspections of the safety related heat exchangers that interface with the "closed-loop" SWS have shown no evidence of clam infestations.

As discussed in our letter to the NRC dated March 4, 1992, (RBG-36584), the SWS, prior to the fourth Refuel Outage (RF-4), was an "open" recirculating system utilizing clarified Mississippi River water as a makeup source. The clarified Mississippi River water provided the source for Asiatic Clam infestation and the "open" SWS design provided a pathway for the larvae and/or clams to infest the SWS and associated safety related equipment.

5.0 #143

L.5 The proposed change updates the referenced ANSI standard to the latest version. The old ANSI standard was developed with limited capacity of the test equipment available. The current equipment available, and the type used at RBS is "state of the art" and much more sensitive than the old gas chromatographs

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.8 - PROCEDURES AND PROGRAMS

TECHNICAL CHANGE - LESS RESTRICTIVE
(continued)

SD #143

used. The older technology required alternating up and down readings and had a much slower response time than the current instrumentation that reads in real time. The 1975 standard required minimum and maximum concentrations to accommodate the slow response times and prevent freon saturation of the charcoal during testing. The 1980 standard eliminated the maximum value. The 1989 standard eliminated the minimum value based on the greater sensitivity of the instrumentation available. All three standards require a limit on the variance. ANSI N510-1989 reflects the latest technologically advanced methods for testing the charcoal and the instruments used at RBS.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.9 - REPORTING REQUIREMENTS

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This reporting requirement is unnecessary since it generally included in the LER requirements to report fuel cladding failures that exceed expected values or that are caused by unexpected factors, i.e., being seriously degraded. Since the criteria identified in 10 CFR 50.73 have been identified as the criteria in the area of degraded boundaries that necessitates reporting, any minor differences are negligible with regard to safety. Therefore, the current reporting requirement is a duplication of the 10 CFR 50.73 reporting requirement and can be deleted.

A.3 This comment number is not used for this station.

A.4 This change provides additional time to obtain calendar year based analyses results which are needed for submittal of this report. Since the report frequency is unchanged from annually, this change is considered administrative.

A.5 This comment number is not used for this station.

A.6 This comment number is not used for this station.

A.7 This comment number is not used for this station.

A.8 This comment number is not used for this station.

A.9 The initial report requirements for the Annual Report and the annual Radiological Environmental Operating Report are being deleted. These initial reports have been submitted, and deleting the discussion surrounding these initial reports are no longer necessary. Therefore, the deletion is purely administrative in nature.

A.10 Recipients for documents sent to the NRC staff are governed by 10 CFR 50.4. Therefore all references in the Technical Specifications to NRC recipients of reports are being deleted. Since 10 CFR 50.4 is the governing requirement, this deletion is considered administrative.

A.11 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.9 - REPORTING REQUIREMENTS

ADMINISTRATIVE
(continued)

A.12 The reporting of challenges to safety and relief valves is revised from an Annual report to a monthly report. Since no change in the details of the reporting are required, this is a change in the timing only and is considered an administrative change.

A.13 The applicable LCOs reference the COLR without these additional cross references. Therefore, the references to the LCOs serve no functional purpose, and their removal is purely an administrative difference in presentation.

A.14 The deletion of the Biofouling Prevention and Detection Program is being proposed based upon a River Bend Station submittal dated January 14, 1994 (RBG-39896). The program is no longer necessary based upon improvements to the service water system.

A.15 The one time exclusion of the result of the misoriented fuel bundle analysis for River Bend is now obsolete. Therefore, its deletion has no impact.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

M.2 A new report is required in conjunction with the changes described in Section 3.4 for the reactor coolant system pressure and temperature limits. In addition, requirements are included for methods used to determine such limits and for submitting the report to the NRC.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.9 - REPORTING REQUIREMENTS

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Details of the methods for implementing this specification are relocated to the ODCM, procedures or the proposed Administrative Controls section of the Technical Specifications as a Routine Report. Additionally, changes to the ODCM, procedures and the Technical Specifications ~~Bases~~ are controlled in accordance with the proposed Technical Specification.

5.0 #57

LA.2 This comment number is not used for this station.

LA.3 Requirements for Special Reports and Startup Reports are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

L.1 This comment number is not used for this station.

L.2 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.10 - RECORD RETENTION

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This comment number is not used for this station.

A.3 This comment number is not used for this station.

A.4 This comment number is not used for this station.

A.5 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 This comment number is not used for this station.

M.2 This comment number is not used for this station.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Record Retention requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

DISCUSSION OF CHANGES
CTS: 6.11 - RADIATION PROTECTION PROGRAM

ADMINISTRATIVE

None in this section.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Radiation Protection Program requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

DISCUSSION OF CHANGES
CTS: 6.12 - HIGH RADIATION AREA

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 This comment number is not used for this station.

LA.2 High Radiation Area control requirements are relocated to the USAR and procedures. Changes to these controls can only be made with pre approval of the NRC consistent with paragraph 20.203 of 10 CFR 20.

"Specific"

L.1 This comment number is not used for this station.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.13 - PROCESS CONTROL PROGRAM

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 This comment number is not used for this station.

A.3 This comment number is not used for this station.

A.4 This comment number is not used for this station.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Process Control Program requirements are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.14 - OFFSITE DOSE CALCULATION MANUAL

ADMINISTRATIVE

A.1 The applicable Specifications provide the requirements without these additional cross references. Therefore, the references to the Specifications serve no functional purpose, and their removal is purely an administrative difference in presentation.

A.2 This comment number is not used for this station.

A.3 This comment number is not used for this station.

A.4 This comment number is not used for this station.

A.5 This comment number is not used for this station.

A.6 Details of these requirements also incorporate changes based upon Generic letter 89-01 and 10 CFR Part 20.

RELOCATED SPECIFICATIONS

None in this section.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Requirements relating to the Offsite Dose Calculation Manual (ODCM) are relocated to the USAR and procedures. Changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

LA.2 This comment number is not used for this station.

LA.3 This comment number is not used for this station.

"Specific"

None in this section.

LAR 93-14R1

DISCUSSION OF CHANGES
CTS: 6.15 - MAJOR CHANGES TO RADIOACTIVE LIQUID, ...

ADMINISTRATIVE

A.1 This comment number is not used for this station.

A.2 These specific issues are covered by a more generic program identified in proposed Sections 5.5 and 5.6 which provides for the definition and implementation of these details. Consequently, removing these issues from the Technical Specifications is considered an administrative change.

A.3 Current Technical Specification Sections 3/4.11 and 3/4.12 are provided with this section for markup purposes only. See the appropriate "RELOCATED" discussion of the specific changes below.

A.4 An statement of applicability of SR 3.0.2 or SR 3.0.3 is needed to maintain the current allowances for surveillance frequency extensions since these SRs are not normally applied to frequencies identified in the Administrative Controls section of the Technical Specifications. Since this change maintains current requirements, it is considered a change of presentation method only.

RELOCATED SPECIFICATIONS

R.1 The radioactive material released in liquid effluents to unrestricted areas is limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.2 The dose or dose commitment to members of the public results from cumulative liquid effluent discharges during normal operation over extended periods and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined radioactive liquid effluent - dose projected value is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the

RELOCATED SPECIFICATIONS
(continued)

NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.3 The liquid radwaste treatment system controls the release of site liquid effluents during normal operational occurrences consistent with 10 CFR Part 50, Appendix A, GDC 60. No loss of primary coolant is involved, neither is an accident condition assumed or implied. Further, the evaluation summarized in NEDO-31466 determined the loss of the system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.4 The dose rate due to radioactive material released in gaseous effluents beyond the site boundary is limited to assure compliance with 10 CFR Part 20. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined gaseous effluent dose rate during normal operation are a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.5 The dose due to noble gases released in gaseous effluents during normal operation over extended periods is limited and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined gaseous effluents dose - noble gas values is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

LAR 93-14R1

RELOCATED SPECIFICATIONS
(continued)

R.6 The dose due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days released in gaseous effluents during normal operation over extended periods is limited and is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined gaseous effluents dose - iodine-131, iodine-133, tritium and radionuclides in particulate form releases is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.7 The Gaseous Radwaste Treatment (offgas) System reduces the activity level of the non-condensible fission product gases from fuel defects removed from the main condenser prior to their release to the environs. The operability of the offgas system is not assumed in the analysis of any DBA or transient. Further, the evaluation summarized in NEDO-31466 determined the system to be a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.8 The Ventilation Exhaust Treatment System reduces radioactive materials in gaseous waste prior to their discharge. The operability of the system is not assumed in the analysis of any DBA or transient. Further, the evaluation summarized in NEDO-31466 determined the system to be a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

DISCUSSION OF CHANGES
CTS: 6.15 - MAJOR CHANGES TO RADIOACTIVE LIQUID, ...

RELOCATED SPECIFICATIONS
(continued)

R.9 The solid radwaste system processes wet radioactive waste and operates in accordance with 10 CFR Part 50, Appendix A, for effluent control. The operability of the system is not assumed in the analysis of any DBA or transient. Further, the evaluation summarized in NEDO-31466 determined that radioactive waste is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.10 The dose due to releases of radioactivity and to radiation from uranium fuel cycle during normal operation over extended periods is limited and is intended to assure compliance with 40 CFR Part 190. These limits are not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined radioactive effluents - total dose value is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.11 The radiological environmental program provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public. This program is not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined that this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

DISCUSSION OF CHANGES
CTS: 6.15 - MAJOR CHANGES TO RADIOACTIVE LIQUID, ...

RELOCATED SPECIFICATIONS
(continued)

R.12 The land use census supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public. This program is not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined that this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

R.13 The interlaboratory comparison program confirms the accuracy of the measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public. This program is not related to protection of the public from any DBA or transient analysis. Further, the evaluation summarized in NEDO-31466 determined that this program is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this function did not satisfy the NRC Interim Policy Statement technical specification screening criteria as documented in the Application of Selection Criteria to the RBS TS and have been relocated to plant documents controlled in accordance with 10 CFR 50.59.

TECHNICAL CHANGES - MORE RESTRICTIVE

None in this section.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Details of the methods for implementing this specification are relocated to the USAR and procedures. The guidance documents which dictate the methods are also identified in the USAR. Additionally, changes to the procedures and the USAR are controlled in accordance with 10 CFR 50.59.

"Specific"

None in this section.

LAR 93-14R1

ATTACHMENT 1C

CTS - PSTS

COMPARISON DOCUMENT

REVISION 1

NO SIGNIFICANT HAZARDS CONSIDERATIONS

LAR 93-14 R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 6.8 - PROCEDURES AND PROGRAMS

5.0 #30

"L2" CHANGE

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would limit the scope of procedures requiring consideration of NUREG-0737 to emergency operating procedures. Even though NUREG-0737 also covers shift turnover, use of overtime, control room access, etc., this is considered reasonable since the intent of NUREG-0737 was to address "emergency operating procedures". These procedures are not considered as initiators for any previously evaluated accident and although they may be required for the mitigation of an evaluated accident, this change is not proposing revisions to these procedures. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to provide for adequate emergency operating procedures.

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 6.8 - PROCEDURES AND PROGRAMS

"L4" CHANGE

5.0 #28

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the program associated with the prevention and detection of Asiatic Clams (Corbicula) based upon improvements to the non-safety related Normal Service Water System (SWS). This program is not considered as an initiator for any previously evaluated accident. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   Preventation of Asiatic Clam infestation to the SWS and associated safety related equipment is ensured by the "closed-loop" design of the SWS. Therefore, the change does not involve a significant reduction in a margin of safety.

LAR 93-14R1

NO SIGNIFICANT HAZARDS CONSIDERATIONS
CTS: 6.8 - PROCEDURES AND PROGRAMS

"L5" CHANGE

5.0 #11.3

Entergy Operations Inc., has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change represents only a change in references used to perform testing and a minor change in testing methodology. The change in testing methodology has no affect on the end results of the test. Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change will continue to provide for adequate testing of charcoal filters.

LAR 93-14R1

ATTACHMENT 2

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

CHAPTER 5 REVISED PAGES

2A: MARKUP OF ITS

2B: DISCUSSION OF CHANGES

LAR 93-14R1

ATTACHMENT 2A

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

MARKUP OF ITS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

B1 5.1.1 The [Plant ~~Superintendent~~] manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

B1 The [Plant ~~Superintendent~~] manager, or his designee, ~~in accordance with approved administrative procedures,~~ shall approve, prior to implementation, each proposed test, ~~or~~ experiment, ~~and proposed~~ or ~~changes and~~ modifications to ~~unit~~ systems or equipment that affect nuclear safety.

C2

5.0 #73

5.1.2 B1 The [Shift ~~Supervisor (SS)~~] superintendent shall be responsible for the control room command function. ~~A management directive to this effect, signed by the [highest level of corporate or site management] shall be issued annually to all station personnel.~~ During any absence of the [SS] from the control room while the unit is in MODE 1, 2, or 3, an individual with ~~a valid~~ an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while the unit is in MODE 4 or 5, an individual with ~~a valid~~ an active SRO license or Reactor Operator (RO) license shall be designated to assume the control room command function.

C44 B1 C23 C23 B1

shift superintendent — 5.0 #73

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.0 #73

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications,

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the ~~[FSAR]~~ USAR;

B1

B1

b. The [plant ~~Superintendent~~ manager] shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

P10

c. ~~The [a~~ A specified corporate executive ~~position]~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and ~~. The specified corporate executive shall be documented in the USAR~~ e

5.0 #73

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall ~~be as follows~~ include the following:

a. ~~Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.~~

INSERT 2A — C24

(continued)

(LAR 93-14R1)

INSERT 2A

A non-licensed operator shall be on site when fuel is in the reactor and an additional non-licensed operator shall be on site while the unit is in MODE 1, 2, or 3.

5.2 Organization

5.2.2 Unit Staff (continued)

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

C24 INSERT 3A

B1

~~c.~~ d. A ~~[~~health physics technician~~]~~ shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

C25 ~~d. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.~~

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).

B1

Adequate shift coverage shall be maintained without routine heavy use of overtime. ~~The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week, while the unit is operating.~~ However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

(continued)

LAR 93-1421

5.0 #77

<u>INSERT 3A</u>

specifications

and 5.22.g

c. Shift crew composition may be one less than the minimum requirement of 10CFR50.54(m)(2)(i) and 5.2.2.a 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.

5.2 Organization

5.2.2 Unit Staff (continued)

[3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

(B1) Any deviation from the above guidelines shall be authorized ~~in advance~~ by the {Plant manager ~~Superintendent}~~ or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the {Plant manager ~~Superintendent}~~ or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.]

~~OR~~

(B1) ~~[The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).]~~

(B1) f. The {Operations superintendent ~~Manager or Assistant Operations Manager}~~ shall hold an SRO license.

(5.0 #73) g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift superintendent ~~Supervisor (SS)~~ in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

(C24) (C26) INSERT 4A

LAR 93-14R1

INSERT 4A

In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition[a]
[Single Unit Facility]

POSITION[b]	MINIMUM CREW NUMBER	
	UNIT IN MODE 1, 2, OR 3	UNIT IN MODE 4 OR 5
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA[c]	1	None

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours to accommodate unexpected absences of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2.2-1. 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.

(b) Table Notation:

SS - [Shift Supervisor] with a Senior Reactor Operator license;
SRO - Individual with a Senior Reactor Operator license;
RO - Individual with a Reactor Operator license;
AO - Auxiliary Operator;
STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift.

LAR 93-14R1

C24

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition(a)
[Two Units With a Common Control Room]
(Totals for Both Units)

POSITION(b)	MINIMUM CREW NUMBER		
	EACH UNIT IN MODE 1, 2, OR 3	ONE UNIT IN MODE 1, 2, OR 3, AND ONE UNIT IN MODE 4 OR 5 OR DEFUELED	EACH UNIT IN MODE 4 OR 5 OR DEFUELED
SS	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA(c)	1	1	None

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours 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 5.2.2-1. 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.

(b) Table Notation:

SS - [Shift Supervisor] with a Senior Reactor Operator license for each unit whose reactor contains fuel.

SRO - Individual with a Senior Reactor Operator license for each unit whose reactor contains fuel. Otherwise, provide an individual for each unit who holds a Senior Reactor Operator license for the unit assigned. During CORE ALTERATIONS on either unit at least one licensed SRO or licensed SRO limited to fuel handling, who has no other concurrent responsibilities, must be present.

RO - Individual with a Reactor Operator license or a Senior Reactor Operator license for unit assigned. At least one RO shall be assigned to each unit whose reactor contains fuel and one RO shall be assigned as relief operator for unit(s) in MODE 1, 2, or 3. Individuals acting as relief operators shall hold a license for both units. Otherwise, for each unit, provide a relief operator who holds a license for the unit assigned.

AO - At least one auxiliary operator shall be assigned to each unit whose reactor contains fuel.

STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift.

LAR 93-14R1

C24

Table 5.2.2-1 (page 1 of 1)
Minimum Shift Crew Composition(a)
[Two Units With Two Control Rooms]
(Numbers for Each Unit)

POSITION(b)	MINIMUM CREW NUMBER			
	UNIT IN MODE 1, 2, OR 3; OTHER UNIT IN MODE 1, 2, OR 3	UNIT IN MODE 4 OR 5; OTHER UNIT IN MODE 1, 2, OR 3	UNIT IN MODE 1, 2, OR 3; OTHER UNIT IN MODE 4 OR 5 OR DEFUELED	UNIT IN MODE 4 OR 5; OTHER UNIT IN MODE 4 OR 5 OR DEFUELED
SS	1(d)	1(d)	1(d)	1(d)
SRO	1	None	1	None
RO	2	1	2	1
AO	2	1	2	2(e)
STA(c)	1(d)	None	1	None

(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 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 5.2.2-1. 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.

(b) Table Notation:

SS - [Shift Supervisor] with a Senior Reactor Operator license;
SRO - Individual with a Senior Reactor Operator license;
RO - Individual with a Reactor Operator license;
AO - Auxiliary Operator;
STA - Shift Technical Advisor.

(c) The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on Shift.

(d) Individual may fill the same position on the other unit if licensed for both.

(e) One of the two required individuals may fill the same position on the other unit.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

B1 ~~[Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.]~~

INSERT 8A

5.3.1 Each member of the ~~unit staff shall meet or exceed the minimum~~ qualifications of ~~[Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~ ~~In addition, the Shift Technical Advisor shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.~~

P2

C26

LAR 93-14R1

INSERT 8A

ANSI/ANS 3.1-1978 for comparable positions, except for the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

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5.0 ADMINISTRATIVE CONTROLS

C27

5.4 Training

5.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the [position title] and shall meet or exceed the requirements and recommendations of Section [ ] of [an ANSI Standard acceptable to the NRC staff] and 10 CFR 55, and, for appropriate designated positions, shall include familiarization with relevant industry operational experience.

LAR 93-14R1

5.0 ADMINISTRATIVE CONTROLS

C28

~~5.5 Reviews and Audits~~

Reviewer's Note: The licensee shall describe the method(s) established to conduct independent reviews and audits. The methods may take a range of forms acceptable to the NRC. These metnods may include creating an organizational unit or a standing or ad hoc committee, or assigning individuals capable of conducting these reviews and audits. When an individual performs a review function, a cross disciplinary review determination is necessary. If deemed necessary, such reviews shall be performed by the review personnel of the appropriate discipline. Individual reviewers shall not review their own work. Regardless of the method used, the licensee shall specify the functions, organizational arrangement, responsibilities, appropriate ANSI/ANS 3.1-1981 qualifications, and reporting requirements of each functional element or unit that contributes to these processes.

Reviews and audits of activities affecting plant safety have two distinct elements. The first element is the reviews performed by plant staff personnel to ensure that day to day activities are conducted in a safe manner. These reviews are described in Section 5.5.1. The second element, described in Section 5.5.2, is the [offsite] reviews and audits of unit activities and programs affecting nuclear safety that are performed independent of the plant staff. The [offsite] reviews and audits should provide integration of the reviews and audits into a cohesive program that provides senior level utility management with an assessment of facility operation and recommends actions to improve nuclear safety and plant reliability. It should include an assessment of the effectiveness of reviews conducted according to Section 5.5.1.

5.5.1 Plant Reviews

Reviewer's Note: The licensee shall describe provisions for plant reviews (organization, reporting, records) and the appropriate ANSI/ANS Standard for personnel qualification.

5.5.1.1 Functions

The [plant review method specified in Specification 5.5.1] shall, as a minimum, incorporate functions that:

a. Advise the [Plant Superintendent] on all matters related to nuclear safety;

(continued)

~~5.5 Reviews and Audits~~ ℮

C28

5.5.1.1 Functions (continued)

b. Recommend to the [Plant Superintendent] approval or disapproval of items considered under Specifications 5.5.1.2.a through 5.5.1.2.e prior to their implementation, except as provided in Specification 5.7.1.3;

c. Determine whether each item considered under Specifications 5.5.1.2.a through 5.5.1.2.d constitutes an unreviewed safety question as defined in 10 CFR 50.59; and

d. Notify the [Vice President—Nuclear Operations] of any safety significant disagreement between the [review organization or individual specified in Specification 5.5.1] and the [Plant Superintendent] within 24 hours. However, the [Plant Superintendent] shall have responsibility for resolution of such disagreements pursuant to Specification 5.1.1.

5.5.1.2 Responsibilities

The [plant review method specified in Specification 5.5.1] shall be used to conduct, as a minimum, reviews of the following:

a. All proposed procedures required by Specification 5.7.1.1 and changes thereto;

b. All proposed programs required by Specification 5.7.2 and changes thereto;

c. All proposed changes and modifications to unit systems or equipment that affect nuclear safety;

d. All proposed tests and experiments that affect nuclear safety; and

e. All proposed changes to these Technical Specifications (TS), their Bases, and the Operating License.

(continued)

~~5.5 Reviews and Audits (continued)~~

C28

5.5.2 [Offsite] Review and Audit

> Reviewer's Note: The licensee shall describe the provisions for reviews and audits independent of the plant's staff (organization, reporting, and records) and the appropriate ANSI/ANS Standards for personnel qualifications. These individuals may be located onsite or offsite provided organizational independence from plant staff is maintained. The [technical] review responsibilities, Specification 5.5.2.4, shall include several individuals located onsite.

5.5.2.1 Functions

The [offsite review and audit provisions specified in Specification 5.5.2] shall, as a minimum, incorporate the following functions that:

a. Advise the [Vice President—Nuclear Operations] on all matters related to nuclear safety;

b. Advise the management of the audited organization, and [its Corporate Management and Vice President—Nuclear Operations], of the audit results as they relate to nuclear safety;

c. Recommend to the management of the audited organization, and its management, any corrective action to improve nuclear safety and plant operation; and

d. Notify the [Vice President—Nuclear Operations] of any safety significant disagreement between the [review organization or individual specified in Specification 5.5.2] and the [organization or function being reviewed] within 24 hours.

5.5.2.2 [Offsite] Review Responsibilities

The [review method specified in Specification 5.6.2] shall be responsible for the review of:

a. The safety evaluations for changes to procedures, equipment, or systems, and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions do not constitute an unreviewed safety question as defined in 10 CFR 50.59;

(continued)

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5.5.2.2 [Offsite] Review Responsibilities (continued)

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

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

d. Proposed changes to TS and the Operating License;

e. Violations of codes, regulations, orders, license requirements, and internal procedures or instructions having nuclear safety significance;

f. All Licensee Event Reports required by 10 CFR 50.73;

g. Plant staff performance;

h. Indications of unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect nuclear safety;

i. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective action to prevent recurrence;

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

k. The performance of the corrective action system.

Reports or records of these reviews shall be forwarded to the [Vice President—Nuclear Operations] within 30 days following completion of the review.

5.5.2.3 Audit Responsibilities

The audit responsibilities shall encompass:

a. The conformance of unit operation to provisions contained within the TS and applicable license conditions;

b. The training and qualifications of the unit staff;

(continued)

C28

~~5.5 Reviews and Audits~~

5.5.2.3 Audit Responsibilities (continued)

c. The implementation of all programs required by Specification 5.7.2;

d. Actions taken to correct deficiencies occurring in equipment, structures, systems, components, or method of operation that affect nuclear safety; and

e. Other activities and documents as requested by the [Vice President—Nuclear Operations].

Reports or records of these audits shall be forwarded to the [Vice President—Nuclear Operations] within 30 days following completion of the review.

5.5.2.4 [Technical] Review Responsibilities

The [technical] review responsibilities shall encompass:

a. Plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources that may indicate areas for improving plant safety;

b. Plant operations, modifications, maintenance, and surveillance to verify independently that these activities are performed safely and correctly and that human errors are reduced as much as practical;

c. Internal and external operational experience information that may indicate areas for improving plant safety; and

d. Making detailed recommendations through the [Vice President—Nuclear Operations] for revising procedures, equipment modifications, or other means of improving nuclear safety and plant reliability.

5.5.3 <u>Records</u>

Written records of reviews and audits shall be maintained. As a minimum these records shall include:

a. Results of the activities conducted under the provisions of Section 5.5;

(continued)

C28

~~5.5 Reviews and Audits~~

5.5.3 Records (continued)

b. Recommendations to the management of the organization being audited;

c. An assessment of the safety significance of the review or audit findings;

d. Recommended approval or disapproval of items considered under Specifications 5.5.1.2.a through 5.5.1.2.e; and

e. Determination whether each item considered under Specifications 5.5.1.2.a through 5.5.1.2.d constitutes an unreviewed safety question as defined in 10 CFR 50.59.

1 < move to 5.5.11 >

C1

5.0 #104

LAR 93-14R1

~~TS Bases Control~~
~~5.6~~
Programs and Manuals
5.5

5.5 Programs and Manuals

5.0 ADMINISTRATIVE CONTROLS

~~5.6~~ Technical Specifications (TS) Bases Control Program

INSERT 16A

C2

5.5.11

~~5.6.1~~ a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed ~~according to Specification 5.5.1~~ C29 5

5.0 #85

~~5.6.2~~ b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1 ~~a.~~ A change in the TS incorporated in the license; or

2 ~~b.~~ A change to the ~~updated~~ U FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59. P5

C2

~~5.6.3~~ c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the U FSAR. P5

do not Specification 5.5.11.b C2

~~5.6.4~~ d. Proposed changes that meet the criteria of ~~(a) or (b)~~ above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71. (e) C2

5.0 #86

LAR 93-14R1

INSERT 16A

This program provides a means for processing changes to the Bases of these Technical Specifications.

5.0 ADMINISTRATIVE CONTROLS

5.~~7~~4 Procedures ~~, Programs, and Manuals~~

~~5.7.1 Procedures~~

5.~~7~~4.1 ~~.1~~ ~~Scope~~

Written procedures shall be established, implemented, and maintained covering the following activities:

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1; ~~as stated in [Generic Letter 82-33];~~ C2 P18

C4
~~c. Security plan implementation;~~

~~d. Emergency plan implementation;~~

5.0 #89

c ~~e~~. Quality assurance for effluent and environmental monitoring;

d ~~f~~. Fire Protection Program implementation; and

e ~~g~~. All programs specified in Specification 5.5. ~~5.7.2~~ C1

C29

~~5.7.1.2 Review and Approval~~

~~Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation and reviewed periodically as set forth in administrative procedures.~~

~~5.7.1.3 Temporary Changes~~

~~Temporary changes to procedures of Specification 5.7.1 may be made provided:~~

~~a. The intent of the existing 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 license on the unit affected; and~~

~~(continued)~~

LAR 93-14R1

5.7 Procedures, Programs, and Manuals

C29 5.7.1.3 Temporary Changes (continued)

c. The change is documented and reviewed in accordance with Specification 5.5.1 and approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures within 14 days of implementation.

Programs and Manuals
5.5

C1 5.5 ~~5.7.2~~ Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained. C6

C30 5.7.2.1 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.

5.7.2.2 Process Control Program (PCP)

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes will be accomplished to ensure compliance with 10 CFR 20, 10 CFR 61, and 10 CFR 71; state regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

Licensee initiated changes to the PCP:

C31

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s), and

2. a determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations.

(continued)

C1 5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

C31 ~~5.7.2.2 Process Control Program (PCP) (continued)~~

~~b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].~~

~~5.7.2.3~~ 5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and

C2

b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs C43 ~~required by specification 5.7.3,~~ and descriptions of the information that should be included in the Annual C2 Radiological Environmental Operating, and Radioactive Effluent Release, reports. ~~required by Specification~~ C43 ~~[5.9.1.3] and Specification [5.9.1.4]~~.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

5.0 #96

1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

2. a determination that the change(s) maintain the levels of radioactive effluent control required by C19 1302 10 CFR 20.~~106~~, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance, ~~by the [review method of Specification 5.5.1] and the~~ including approval of the ~~[Plant Superintendent]~~ radiation protection manager; and C28

B1

5.0 #73

(continued)

LAR 93-14R1

C1

5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

~~5.7.2.3~~ 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

~~5.7.2.4~~ 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~[~~the Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, ~~hydrogen recombiner,~~ process sampling, and Standby Gas Treatment~~]~~ Systems. The program shall include the following: (B1)

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

~~5.7.2.5 In Plant Radiation Monitoring~~

~~This program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

C33

~~a. Training of personnel;~~

~~b. Procedures for monitoring; and~~

~~c. Provisions for maintenance of sampling and analysis equipment.~~

(continued)

~~5.7.2 Programs and Manuals (continued)~~

~~5.7.2.6~~ 5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

~~5.7.2.7~~ 5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

P24 ten times the concentration values in

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 CFR 20, Appendix B, Table ~~II,~~ 2 C9 Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.~~106~~ 1302 C19 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~~ the P7 unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

(continued)

5.5
~~5.7~~ ~~Procedures,~~ Programs, and Manuals

5.5.4
~~5.7.2.7~~ Radioactive Effluent Controls Program (continued)

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

P25 INSERT 22A

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary~~, conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;~~

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from ~~each~~ the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; P7

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from ~~each~~ the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and P7

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.

C32

~~5.7.2.8 Radiological Environmental Monitoring Program~~

~~This program is for monitoring the radiation and radionuclides in the environs of the plant. The program shall provide representative measurements of radioactivity in the highest potential exposure pathways and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall be contained in the ODCM,~~

(continued)

LAR 93-14R1

as follows:

INSERT 22A

1. For noble gases: ~~Less than or equal to~~ ≤ 500 mrem/yr to the total body and ~~less than or equal to~~ ≤ 3000 mrem/yr to the skin, and

2. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half-lives ~~greater than eight~~ > 8 days: ~~less than or equal to~~ ≤ 1500 mrem/yr to any organ;

5.0 #99

~~5.7~~ 5.5 ~~Procedures,~~ Programs, and Manuals

C32

5.7.2.8 Radiological Environmental Monitoring Program (continued)

shall conform to the guidance of 10 CFR 50, Appendix I, and shall 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 an 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.

~~5.7.2.9~~ 5.5.5 Component Cyclic or Transient Limit

5.0 #101

B1

This program provides controls to track the ~~FSAR, Section [ ],~~ identified on USAR Table 3.9B-22 cyclic and transient occurrences to ensure that ~~components are~~ the reactor vessel is maintained within the design limits.

P31

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5.7.2.10 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

C34

5.7.2.11 Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

(continued)

5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

5.7.2.11 Inservice Inspection Program (continued)

a. Provisions that inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with ASME Boiler and Pressure Vessel Code and Addenda, Section XI, as required by 10 CFR 50.55a;

b. Th[illegible]visions of SR 3.0.2 are applicable to the frequencies f[illegible]forming inservice inspection activities;

c. An inservice inspection program for piping identified in NRC Generic Letter 88-01 in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01, or in accordance with alternate measures approved by the NRC staff; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

C34

~~5.7.2.12~~ 5.5.6 Inservice Testing Program

SD #104

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

C34 a. Provisions that inservice testing of ASME Code Class 1, 2, and 3 pumps, valves, and snubbers shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a; e

~~b.~~ a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

(continued)

C1

LAR 93-14R1

5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

~~5.7.2.12~~ 5.5.6 Inservice Testing Program (continued)

5.0 #104

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b ~~c.~~ The provisions of SR 3.0.2 are applicable to the above required frequencies for performing inservice testing activities;

c ~~d.~~ The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d ~~e.~~ Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.0 #104

~~5.7.2.13~~ 5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in ~~[~~Regulatory Guide 1.52~~]~~, Revision 2, ~~and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1]~~. and

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5.0 #105

(continued)

C1

5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

~~5.7.2.13~~ 5.5.6 Ventilation Filter Testing Program (VFTP) (continued)

B1

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-1989] at the system flowrate specified below [± 10%]:

ESF Ventilation System	Flowrate
SGTS	12,500 cfm
FBVS	10,000 cfm
CRFAS	4,000 cfm

5.0 #143

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < ~~[0.5]~~ 0.05% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-1989] at the system flowrate specified below [± 10%]:

ESF Ventilation System	Flowrate
SGTS	12,500 cfm
FBVS	10,000 cfm
CRFAS	4,000 cfm

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with ~~[ASTM D3803-1989] at a temperature of ≤ [30°C]~~ Regulatory Guide 1.52, Revision 2, Regulatory Position C.6.a and greater than or equal to the relative humidity specified below:

ESF Ventilation System	Penetration	RH
SGTS	0.175%	70%
FBVS	0.175%	70%
CRFAS	0.175%	70%

(continued)

C1

5.5
~~5.7~~ ~~Procedures,~~ Programs, and Manuals

5.0 #104 ~~5.5.7~~ ~~5.7.2.13~~ Ventilation Filter Testing Program (VFTP) (continued)

B1

~~Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).~~

~~Safety factor = [5] for systems with heaters.~~
~~= [7] for systems without heaters.~~

B1

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ~~ASME~~ ANSI N510-1989] at the system flowrate specified below [± 10%]: 5.0 #143

ESF Ventilation System	Delta P	Flowrate
SGTS	< 8" WG	12,500 cfm
FBVS	< 8" WG	10,000 cfm
CRFAS	< 8" WG	4,000 cfm

e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below ~~[± 10%]~~ when tested in accordance with [~~ASME~~ ANSI N510-1989]:] 5.0 #143

ESF Ventilation System	Wattage
SGTS	≥ 61 kw
FBVS	≥ 49 kw
CRFAS	23 ± 2.3 kw

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.0 #104 ~~5.7.2.14~~ 5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the ~~[Waste Gas Holdup System], [the quantity of radioactivity contained in gas storage tanks or fed into the~~ main Condenser offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks]. ~~The gaseous radioactivity quantities shall be determined following the~~ C21

(continued)

5.5
~~5.7~~ ~~Procedures,~~ Programs, and Manuals

5.0
#104

~~5.7.2.14~~
5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

C21

~~methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].~~

The program shall include:

C21

B1

main condenser offgas treatment system

a. The limits for concentrations of hydrogen ~~and oxygen~~ in the ~~[Waste Gas Holdup System]~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

~~b. A surveillance program to ensure that the quantity of radioactivity contained in [each gas storage tank and fed into the offgas treatment system] is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of [an uncontrolled release of the tanks' contents]; and~~

P8

radioactive material contained in any unprotected outdoor tank is limited to ≤ 10 curies, excluding tritium and dissolved or entrained noble gases.

B8

b ~~c~~. A surveillance program to ensure that the ~~quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the [Liquid Radwaste ~~Treatment~~ System] is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.~~

2 C9

5.0
#110

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

(continued)

C1

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5.5 ~~5.7~~ ~~Procedures,~~ Programs, and Manuals

~~5.7.2 Programs and Manuals (continued)~~ e

5.0 #104

~~5.7.2.16~~ 5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

5.0 #112

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

of the new fuel oil

P19

b. Other properties for ASTM 2D fuel oil are within limits within ~~30~~ 31 days ~~following sampling and~~ of addition to storage tanks; and

31 C11

P9

5.0 #116

c. Total particulate concentration of the fuel oil in the storage tanks is ~~≠~~ ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

P19

C35

~~5.7.2.16 Fire Protection Program~~

~~This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.~~ e

5.0
#104

5.5 Programs and Manuals

5.0 ADMINISTRATIVE CONTROLS

5.5.10 ~~5.8~~ Safety Function Determination Program (SFDP)

~~5.8.1~~ C12 This program ensures loss of safety function is detected and appropriate actions taken. Upon ~~failure to meet two or more LCOs at the same time~~ entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

~~5.8.2~~ The SFDP shall contain the following:

a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

~~5.8.3~~ A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

C46

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable ~~(Case A)~~; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable ~~(Case B)~~; or

(continued)

C1

5.0 #104

5.5 Programs and Manuals
~~5.8 SFDP~~

5.5.10 SFDP
~~5.8.2~~ (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable [redacted].

C46

~~Generic Example:~~

~~Division A~~		~~Division B~~	
~~System i~~		~~System i~~	~~←Case C~~
↓		↓	
~~System ii~~	~~←(Support System Inoperable)~~	~~System ii~~	
↓		↓	
~~System iii~~		~~System iii~~	~~←Case A~~
↓		↓	
~~System iv~~		~~System iv~~	~~←Case B~~

[redacted] The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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C1

5.0 ADMINISTRATIVE CONTROLS

5.6 ~~5.9~~ Reporting Requirements

C37 ~~5.9.1 Routine Reports~~

The following reports shall be submitted in accordance with 10 CFR 50.4.

~~5.9.1.1 Startup Report~~

~~A summary report of plant startup and power escalation testing shall be submitted following:~~

~~a. Receipt of an Operating License;~~

~~b. Amendment to the license involving a planned increase in power level;~~

~~c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier; and~~

~~d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.~~

C36

~~The initial Startup Report shall address each of the startup tests identified in FSAR, Chapter [14], 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 required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and modifications.~~

~~Startup Reports shall be submitted within 90 days following completion of the Startup Test Program; 90 days following resumption or commencement of commercial power operation; or 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 operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.~~

(continued)

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C1

5.6
~~5.9~~ Reporting Requirements

C37

~~5.9.1 Routine Reports (continued)~~

~~5.9.1.2~~
5.6.1 ~~Annual Reports~~ The Occupational Radiation Exposure Report C2

B1

~~NOTE~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections 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 submitted by March 31 of each year. ~~[The initial report shall be submitted by March 31 of the year following initial criticality.]~~ C2 B1

~~Reports required on an annual basis include:~~ C2

C2 ~~a.~~ Occupational Radiation Exposure Report

C19 , for whom monitoring was required,

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance ~~(describe maintenance)~~, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.~~407~~ 2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions~~; and~~. C2

C19 B1

~~[b. Any other unit unique reports required on an annual basis.]~~

(continued)

(LAR 93-1481)

C1

5.6
~~5.9~~ Reporting Requirements

C37

~~5.9.1 Routine Reports (continued)~~

~~5.9.1.3~~
5.6.2 Annual Radiological Environmental Operating Report

B1

~~-----NOTE-----~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.~~
~~----------~~

5.0 #142

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May ~~15~~ of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

B1

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. ~~[The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.]~~ In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)

C1

5.6 ~~5.9~~ Reporting Requirements

C37 ~~5.9.1 Routine Reports (continued)~~

~~5.9.1.4~~ 5.6.3 Radioactive Effluent Release Report

B1

~~NOTE~~
~~A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.~~

C10 INSERT 35A

The Radioactive Effluent Release Report covering the operation of the unit ~~shall be submitted in accordance with 10 CFR 50.36a.~~ The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and ~~P~~rocess ~~C~~ontrol ~~P~~rogram and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.0 #53

C37 ~~5.9.1.5~~ 5.6.4 Monthly Operating Reports

B1 P21

Routine reports of operating statistics and shutdown experience~~[~~, including documentation of all challenges to the main steam safety/relief valves,~~]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

C37 ~~5.9.1.6~~ 5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~[The individual specifications that address core operating limits must be referenced here.]~~

1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR
2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR),
3) LCO 3.2.3, Linear Heat Generation Rate (LHGR), and
4) LCO 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoints.

5.0 #123

C2

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents~~:~~

(continued)

LAR 93-14R1

INSERT 35A

during the previous calendar year shall be submitted by May 1 of each year.

C1

5.6 ~~5.9~~ Reporting Requirements

C37 ~~5.9.1.6~~ 5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

B1 INSERT 36A

~~[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

C37 ~~5.9.1.7~~ 5.6.6 ~~Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~

~~a. The RCS pressure and temperature limits, including heatup ~~and~~ ~~cooldown rates~~, criticality, and hydrostatic and leak test limits, shall be established and documented in the PTLR. [The individual Specifications that address the reactor vessel pressure and temperature limits and the heatup and cooldown rates may be referenced.]~~ P32 C10 B1

~~b. The analytical methods used to determine the pressure and temperature limits ~~including the heatup and cooldown rates~~ shall be those previously reviewed and approved by the NRC in [Topical Report(s), number, title, date, and NRC staff approval document, or staff safety evaluation report for a plant specific methodology by NRC letter and date].~~ P32

~~c. The reactor vessel pressure and temperature limits, ~~including those for heatup and cooldown rates,~~ shall be determined so that all applicable limits ~~(e.g., heatup limits, cooldown limits, and inservice leak and hydrostatic testing limits)~~ of the analysis are met.~~ C10 P32

~~d. The PTLR, including revisions or supplements thereto, shall be provided upon issuance for each reactor vessel fluency period to the NRC.~~ C10

~~Regulatory Guide 1.99, Revision 2 and 10CFR50, Appendices G and H.~~ B1

5.0 #127

(continued)

LAR 93-14R1

INSERT 36A

1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version);

5.6 ~~5.9~~ Reporting Requirements (continued)

5.9.2 Special Reports

C37

[Special Reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit, and their preparation and submittal are designated in the Technical Specifications.]

a

Special Reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

The following Special Reports shall be submitted:

C38

a. In the event an ECCS is actuated and injects water into the RCS in MODE 1, 2, or 3, a Special Report shall be prepared and submitted 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.

C39

[b. If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

C40

c. When a Special Report is required by Condition B or G of LCO 3.3.[3.1], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

C1

5.6
~~5.9~~ Reporting Requirements

5.9.2 Special Reports (continued)

B1

[d. Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.]

5.0 ADMINISTRATIVE CONTROLS

5.10 Record Retention

5.10.1 The following records shall be retained for at least 3 years:

a. All License Event Reports required by 10 CFR 50.73;

b. Records of changes made to the procedures required by Specification 5.7.1.1; and

c. Records of radioactive shipments.

5.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time intervals 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;

C41 c. Records of surveillance activities, inspections, and calibrations required by the Technical Specifications (TS) [and the Fire Protection Program];

d. Records of sealed source and fission detector leak tests and results; and

e. Records of annual physical inventory of all sealed source material of record.

5.10.3 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 FSAR;

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;

(continued)

5.10 Record Retention

5.10.3 (continued)

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 [FSAR, Section X];

f. Records of reactor tests and experiments;

g. Records of training and qualification for members of the unit staff;

h. Records of inservice inspections performed pursuant to the TS;

i. Records of quality assurance activities required by the Operational Quality Assurance (QA) Manual [not listed in Specification 5.10.1 and which are classified as permanent records by applicable regulations, codes, and standards];

j. Records of reviews performed for changes made to procedures, equipment, or reviews of tests and experiments pursuant to 10 CFR 50.59;

C41

k. Records of the reviews and audits required by Specification 5.5.1 and Specification 5.5.2;

l. Records of the service lives of all hydraulic and mechanical snubbers required by [document where snubber requirements relocated to], including the date at which the service life commences, and associated installation and maintenance records;

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 (these records should include procedures effective at specified times and QA records showing that these procedures were followed);

n. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program; and

[o. Records of pre-stressed concrete containment tendon surveillances.]

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5.0 #133

B1 [High Radiation Area]
[5.11] 7

5.0 ADMINISTRATIVE CONTROLS

[5.~~11~~ 7 High Radiation Area]

5.~~11~~ 7.1 Pursuant to 10 CFR 20, paragraph 20.~~203(c)(5)~~ 1601(c), in lieu of the requirements of 10 CFR 20.~~203(c)~~ 1601(a), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., [Health Physics Technicians]) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

5.0 #73

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

B1

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~[Radiation Protection Manager]~~ health physics supervision in the RWP.

5.0 #73

5.~~11~~ 7.2 In addition to the requirements of Specification 5.~~11~~ 7.1, areas with radiation levels ≥ 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the ~~Shift Foreman~~ shift supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel

(continued)

5.~~11~~ 7.2 (continued)

under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

(B1)

5.~~11~~ 7.3 In addition to the requirements of Specification 5.7.1, for individual high radiation areas with radiation levels of ~~>~~ ≥ 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

LAR 93-14R1

ATTACHMENT 2B

ITS - PSTS

COMPARISON DOCUMENT

REVISION 1

DISCUSSION OF CHANGES

LAR 93-14R1

DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

BRACKETED ADMINISTRATIVE CHOICE

B.1 Brackets removed and optional wording preferences revised to reflect appropriate plant specific requirements.

PLANT SPECIFIC DIFFERENCE

P.1 This comment number is not used for this station.

P.2 Plant specific staff qualifications have been previously reviewed and approved as identified in NUREG-1172, as supplemented.

P.3 This comment number is not used for this station.

P.4 This comment number is not used for this station.

P.5 The safety analysis report for this station is identified as the Updated Safety Analysis Report and is correctly referred to as the USAR.

P.6 This comment number is not used for this station.

P.7 The references to "each" unit and "on the unit affected" have been revised to reflect that only one unit is located at this site.

P.8 This requirement is only applicable to PWRs.

P.9 The current plant specific requirement details for diesel generator fuel oil testing are retained in the proposed program.

P.10 Where possible, plant specific management position titles in the proposed Technical Specifications are replaced with generic titles as provided in ANSI/ANS 3.1. Personnel who fulfill these positions are required to meet specific qualifications as detailed in proposed Specification 5.3, and compliance details relating to the plant specific management position titles are identified in licensee controlled documents (such as the USAR). The two major specific replacements are the generic "plant manager" for the manager level individual responsible for the overall safe operation of the plant and the generic descriptive use of "the corporate executive responsible for overall plant nuclear safety" in place of the Vice President position. The plant specific titles fulfilling the duties of these generic positions will continue to be defined, established, documented and updated in a plant controlled document with specific regulatory review requirements for changes (such as the USAR).

DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

PLANT SPECIFIC DIFFERENCE
(continued)

This approach is consistent with the intent of Generic Letter 88-06 which recommended, as a line item improvement, relocation of the corporate and unit organization charts to licensee controlled documents. The intent of the Generic Letter, and of this proposed change, is to reduce the unnecessary burden on NRC and licensee resources being used to process changes due solely to personnel titles changes during reorganizations. Since this change does not eliminate any of the qualifications, responsibilities or requirements for these personnel or the positions, the change is considered to be a change in presentation only and is therefore administrative.

P.11 This comment number is not used for this station.

P.12 This comment number is not used for this station.

P.13 This comment number is not used for this station.

P.14 This comment number is not used for this station.

P.15 This comment number is not used for this station.

P.16 This comment number is not used for this station.

P.17 This comment number is not used for this station.

P.18 The reference to Generic Letter (GL) 82-33 has been deleted since this GL does not cover all requirements of NUREG - 0737 as presently reflected in the current Technical Specifications.

P.19 Appropriate clarification has been provided for sampling of diesel fuel oil.

P.20 This comment number is not used for this station.

P.21 The discussion has been modified to more accurately describe the safety/relief valves being discussed.

P.22 This comment number is not used for this station.

P.23 This comment number is not used for this station.

P.24 This change has been made to clarify that the limitations on the concentrations of radioactive material released in effluents to unrestricted areas, conform to ten times the concentration values in 10CFR 20, Appendix B.

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DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

PLANT SPECIFIC DIFFERENCE
(continued)

P.25 Plant specific criteria has been provided for limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary.

P.26 This comment number is not used for this station.

P.27 This comment number is not used for this station.

P.28 This comment number is not used for this station.

P.29 This comment number is not used for this station.

P.30 This comment number is not used for this station.

P.31 The plant specific requirements for the component cyclic or transient limit program are identified.

This comment number is not used for this station.

5.0 #127

P.32 ~~The PTLR discussion is modified to reflect the retention of requirements in the associated LCO and editorial changes are made to reduce the repetition of information. As part of the retention of specific limits in ITS SRs 3.4.11.8 and 3.4.11.9 the acceptance criteria for these SRs were changed to include the limit (e.g., less than or equal to 100°F/50°F). This was an administrative change in presentation only to provide consistency in the limits identified and resulted in no real change to the limits.~~ e

CHANGE/IMPROVEMENT TO NUREG STS

C.1 The sections related to Technical Specification Bases Control, Procedures, Programs and Manuals, and the Safety Function Determination Program have been reformatted to include all programs in the Programs and Manuals section. Additionally, the Procedures requirements have been separated into an individual section apart from the Programs and Manuals. Appropriate renumbering of these sections and the ones that follow them are also incorporated, along with revisions to references to these sections to reflect these changes and the relocation of requirements to licensee control conforming with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.2 This change reflects an editorial correction to the wording of NUREG-1434.

C.3 This comment number is not used for this station.

DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

C.4 Procedures to implement the Emergency Plan and the Security Plan are required by 10 CFR 50, Appendix E and 10 CFR 50.54(p). Therefore, there is no need to repeat the requirements. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.5 This comment number is not used for this station.

C.6 This change is to provide consistency of requirements for the items specified as Manuals. This change prevents potential confusion and misinterpretation.

C.7 This change editorially rewords the requirement. The change avoids unnecessary confusion that may develop in attempting to determine the intent of the wording.

C.8 This comment number is not used for this station.

C.9 10 CFR 20, Appendix B, has been revised such that the correct reference is Table 2 rather than Table II.

C.10 The change avoids unnecessary confusion that may develop in attempting to determine the intent of different wording.

C.11 This change is made to provide consistency with the generic surveillance frequencies.

C.12 The description of the entry conditions into the SFDP are clarified and generalized to assure that they include all possible required entry conditions.

C.13 This comment number is not used for this station.

C.14 This comment number is not used for this station.

C.15 This comment number is not used for this station.

C.16 This comment number is not used for this station.

C.17 This comment number is not used for this station.

C.18 This comment number is not used for this station.

C.19 These changes provide for consistency with the new 10 CFR 20.

C.20 This comment number is not used for this station.

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DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

C.21 This change removes a requirement that is only applicable to Pressurized Water Reactors (PWRs).

C.22 This comment number is not used for this station.

C.23 These changes reflect editorial corrections to the wording of NUREG-1434. These changes conform with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.24 The requirements of Table 5.2.2-1 are removed for the Technical Specifications and will be controlled by the licensee's administrative controls. 10 CFR 50.54 provides the requirements for shift complement regarding licensed operators. Additionally, Section 5.2.2.b specifies when a licensed operator must be in the control room. The Table 5.2.2-1 requirements associated with the auxiliary (non-licensed) operators are retained as 5.2.2.a and 5.2.2.c retains the allowance for unexpected absences.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993 with editorial rewording of 5.2.2.a to be consistent with the wording of 5.2.2.b, the addition of the allowance for unexpected absences of an auxiliary operator which was allowed on Table 5.2.2-1, and the addition of the allowance for an on shift SS or SRO to perform the STA function as was allowed by Table 5.2.2-1.

C.25 The requirements for SRO presence during fuel handling and core alterations are contained in 10 CFR 50.54. Therefore, there is no need to repeat the requirements. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.26 The requirements for the STA to meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift is moved from 5.3.1 to 5.2.2.g. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.27 The retraining and replacement training program requirements for the unit staff are relocated. Details of this program can be adequately controlled by the licensee's administrative controls. Technical Specification Section 5.3, Unit Staff

DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

Qualifications, provides adequate requirements to assure an acceptable, competent unit staff. In accordance with Technical Section 5.3 and RBS commitments, members of the unit staff meet or exceed the minimum qualifications of the specific Regulatory Guides or ANSI Standards for their positions. Additionally, Technical Specification Section 5.2, Organization, details unit staff requirements. Sections 5.2.2.a and 5.2.2.b and 10 CFR 50.54 describe the minimum shift crew composition and delineates those positions which require an RO or SRO license. Training and qualification for these positions are specified in 10 CFR 55.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.28 The review and audit requirements can be adequately controlled by the licensee's administrative controls such as the USAR and station administrative procedures. This results in an equivalent level of regulatory authority while providing for a more appropriate change control process. The net affect of the change is the level of safety of facility operation is unaffected and NRC and utility resources associated with processing license amendments to this Administrative Control are optimized.

The onsite review function, composition, alternate membership, meeting frequency, quorum, responsibilities, authority and records are all covered in equivalent detail in ANSI N18.7-1976 and the current Technical Specification requirements are contained in Section 13.4 of the USAR.

The offsite review group is also addressed, although with less detail, in ANSI N18.7-1976. Section 13.4 of the USAR includes the requirements for the offsite review group.

Audit requirements are specified in the QA Program to satisfy 10 CFR 50, Appendix B, Criterion XVIII. These audit requirements are augmented by the inclusion in Section 13.4 of the USAR those audit requirements currently contained in the Technical Specifications. In addition, audits are also covered by ANSI N18.7, ANSI N45.2, 10 CFR 50.54(t), 10 CFR 50.54(p), and 10 CFR 73. Therefore, duplication of the requirements contained in the above documents by the Administrative Controls Section of the Technical Specifications does not enhance the level of nuclear safety for the unit. Therefore, the provisions relating to audits are not necessary to assure safe operation of the facility.

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993. References to these controls are corresponding moved to the license's administrative controls.

C.29 The details of review and approval requirements for procedures and programs can be adequately controlled by the licensee's administrative controls. These requirements are included in Section 13.4 of the USAR. This proposal is based on the existence of the following requirements which are duplicative of 10 CFR 50.36 in these areas and which assure operation of the facility in a safe manner. The requirement for procedures is mandated by 10 CFR 50, Appendix B, Criterion II (Second sentence) and Criterion V. ANSI N18.7-1976, which is an NRC Staff-endorsed document used in the development of many licensee QA plans, also contains specific requirements related to procedures.

ANSI N18.7-1976, Section 5.2.2 discusses Procedure Adherence. This section clearly states that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. ANSI N18.7-1976 also discusses temporary changes to procedures, and requires review and approval of procedures to be defined.

ANSI N18.7-1976, Section 5.2.15 describes the review, approval and control of procedures. The section describes the requirements for the Licensee's Quality Assurance program to provide measures to control and coordinate the approval and issuance of documents, including changes thereto, which prescribe all activities affecting quality. The Section further states that each procedure shall be reviewed and approved prior to initial use. The reviews required are also described.

Licensees can continue to implement the requirements of 10 CFR 50, Appendix B, regarding procedures without duplicating the necessity of procedure requirements in the facility Technical Specifications. Safe operation of the plant will continued to be maintained, and therefore, the requirements for procedures and their control should not be re-addressed in Technical Specifications. Duplication of the provisions related to procedures is not necessary to assure safe operation of the facility.

Procedure control requirements are supplemented by current Technical Specification requirements in the USAR. This change results in an equivalent level of regulatory authority while

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

providing for a more appropriate change control process. The net affect of the change is the level of safety of facility operation is unaffected and NRC and utility resources associated with processing license amendments to this Administrative Control are optimized.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.30 The requirement for procedures to implement the personnel radiation protection requirements of 10 CFR 20 can be adequately controlled by the licensee's administrative controls. These procedures are developed to ensure nuclear plant personnel safety and have no impact on nuclear safety. Additionally, nuclear plant personnel are not "members of the public." Thus, the principal operative standard in Section 182a of the Atomic Energy Act; "health and safety of the public" does not apply. Based on these considerations, the Radiation Protection Program administrative control is not necessary to assure operation of the facility in a safe manner and can be deleted from Technical Specifications. The requirement to have procedures to implement Part 20 is also contained within 10 CFR 20.1101(b). Periodic review of these procedures is addressed under 10 CFR 20.1101(c).

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.31 The requirement for a Process Control Program (PCP) and its review and approval requirements can be adequately controlled by the licensee's administrative controls. The PCP can be adequately described in another controlled documents, e.g., the ODCM and the USAR. Control of changes is preserved by 10 CFR 50.54a. The PCP implements the requirements of 10 CFR 20, 10 CFR 61, and 10 CFR 71. Relocating the description of the PCP does not affect the safe operation of the facility. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.32 The radiological environmental monitoring program is required to be part of the ODCM, the additional details of the program can be adequately controlled by the licensee's administrative controls. The ODCM requirements in conjunction with the USAR

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DISCUSSION OF CHANGES TO NUREG-1434
CHAPTER 5 - ADMINISTRATIVE CONTROLS

CHANGE/IMPROVEMENT TO NUREG STS
(continued)

will continue to control this program. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.33 The requirement for an in plant radiation monitoring program and the associated details of the program can be adequately controlled by the licensee's administrative controls. This program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program was developed to minimize radiation exposure to plant personnel post-accident and has no impact on nuclear safety. Additionally, nuclear plant personnel are not "members of the public." Thus, the principal operative standard in Section 182a of the Atomic Energy Act; "health and safety of the public" does not apply. Based on these considerations, the In Plant Radiation Monitoring Program administrative control is not necessary to assure operation of the facility in a safe manner and can be deleted from Technical Specifications. This program will continue to be controlled by the licensee's administrative controls. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.34 The requirement for an Inservice Inspection Program and the Inservice Testing Program can be adequately controlled by the licensee's administrative controls and the requirements of 10 CFR 50.55a. Removal of the Inservice Inspection Program and the details of the Inservice Testing Program conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

S.O #116

INSERT 9A

C.35 ~~This comment number is not used for this station.~~

C.36 The requirement for a Startup Report and the associated details of the report and when it should be submitted can be adequately controlled by the licensee's administrative controls. The report was a summary of plant startup and power escalation testing following receipt of the Operating License, increase in licensed power level, installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. The report provided a mechanism for NRC to review the appropriateness of licensee activities after-the-fact, but provided no regulatory authority once the report was submitted (i.e., no requirement

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C.35 The requirement for a Fire Protection Program and the associated details of the program can be adequately controlled by the licensee's administrative controls. The Fire Protection Program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire. The administrative control provides assurance that the capability to provide for alternate/ dedicated safe shutdown in accordance with 10 CFR 50, Appendix R. As such, it does not directly assure nuclear safety, but rather allows for the ability to place the unit in a more stable condition in the event of a fire.

The deletion of this administrative control from Technical Specifications is also consistent with the guidance in NRC Generic Letter 86-10 "Implementation of Fire Protection Requirements." In that letter, the NRC concluded the provisions of 10 CFR 50.59 should apply directly to changes the licensee desired to make in the fire protection program so long as those changes did not adversely affect the ability to achieve and maintain safe shutdown. The standard license condition, included within 86-10, stated that changes which adversely affected the ability to achieve and maintain safe shutdown in the event of a fire required prior approval of the Commission. Thus, the license condition established as part of the NRC Generic Letter 86-10 implementation also makes this administrative control unnecessary.

Based on these considerations, the Fire Protection Program administrative control is not necessary to assure operation of the facility in a safe manner and can be deleted from Technical Specifications.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

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for Commission approval). The approved 10 CFR 50, Appendix B, Quality Assurance Plan and Startup Test Program (USAR Section 13) provide assurance the listed activities are adequately performed and that appropriate corrective actions, if required, are taken.

Given that the report was required to be provided to the Commission no sooner than 90 days following completion of the respective milestone, it was clearly not necessary to assure operation of the facility in a safe manner for the interval between completion of the startup testing and submittal of the report. Additionally, given there is no requirement for the Commission to approve the report, then the Startup Report is not necessary to assure operation of the facility in a safe manner.

Based on these considerations, the Start-up Report may be removed from TS and relocated to a licensee controlled document.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.37 Since no report descriptions remain in the Special Report sections this section heading is removed, the corresponding Routine Report Section designation is removed, and the items are renumbered to reflect these changes.

C.38 The requirement for a Special Report following an ECCS injection and the associated details of the report and when it should be submitted can be adequately controlled by the licensee's administrative controls. Title 10, Part 50, Section 73 already provides the requirement for the license to submit a Licensee Event Report in the event of an ECCS actuation. The report is required to be submitted within 30 days and will contain the same type of information as the special report. Removing the duplicative requirement from TS has no impact on assuring safe operation of the facility since the requirement to submit a report to Commission still exists in 10 CFR 50.73.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.39 The requirement for a Special Report following EDG failures, the associated details of the report, and when it should be

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(continued)

submitted can be adequately controlled by the licensee's administrative controls. The report provided a mechanism for NRC to review the appropriateness of licensee activities after-the-fact, but provided no regulatory authority once the report was submitted (i.e., no requirement for Commission approval). Given that the report was required to be provided to the Commission no sooner than 30 days following the EDG failure, it was clearly not necessary to assure operation of the facility in a safe manner for the interval between the EDG failure and submittal of the report. Additionally, given there is no requirement for the Commission to approve the report, then this Special Report is not necessary to assure operation of the facility in a safe manner.

Based on these considerations, this Special Report may be removed from TS and relocated to a licensee controlled document.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.40 The requirement for a Special Report following extended Post Accident Monitoring instrumentation inoperability and the associated details of the report and when it should be submitted can be adequately controlled by the licensee's administrative controls. The report provided a mechanism for NRC to review the appropriateness of licensee activities after-the-fact, but provided no regulatory authority once the report was submitted (i.e., no requirement for Commission approval). Given that the report was required to be provided to the Commission no sooner than 30 days following the instrumentation failure, it was clearly not necessary to assure operation of the facility in a safe manner for the interval between failure of the instrumentation and submittal of the report. Additionally, given there is no requirement for the Commission to approve the report, then this Special Report is not necessary to assure operation of the facility in a safe manner.

Based on these considerations, this Special Report may be removed from TS and relocated to a licensee controlled document. This information has been added to the bases for the LCO Required Actions which required the Special Report to be written. This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

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C.41 Record retention requirements and the associated details of the report can be adequately controlled by the licensee's administrative controls. The requirements on record retention may be deleted from Technical Specifications on the basis that they are adequately addressed by the QA Plan (10 CFR 50, Appendix B, Criteria XVII), that the current Technical Specification specific requirements will be included in the USAR, and because provisions relating to record keeping do not assure operation of the facility in a safe manner.

Facility operations are performed in accordance with approved written procedures. Areas include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, safety related maintenance, surveillance and testing, and radiation control. Facility records document appropriate station operations and activities. Retention of these records provides documentation retrievability for review of compliance with requirements and regulations. Post compliance review of records does not assure operation of the facility in a safe manner as activities described in these documents have already been performed. Numerous other regulations such as 10 CFR 20, Subpart L, 10 CFR 50.71, etc. also require the retention of certain records related to operation of the nuclear plant.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

C.42 The accepted alternate methods of complying with 10 CFR 20.203(c) for high radiation areas can only be changed with prior NRC approval and can be adequately controlled by the licensee's administrative controls. These controls are developed to ensure nuclear plant personnel safety and have no impact on nuclear safety. Additionally, nuclear plant personnel are not "members of the public." Thus, the principal operative standard in Section 182a. of the Atomic Energy Act; 'health and safety of the public' does not apply. Based on these considerations, the Radiation Protection Program administrative control is not necessary to assure operation of the facility in a safe manner and can be deleted from Technical Specifications.

This change conforms with the changes suggested by the NRC in the letter W. T. Russell (NRC) to the Chairpersons of the Owners' Groups Technical Specifications Committees dated October 25, 1993.

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5.0 #123

This comment number is not used for this station

C.43 ~~In general the format of the NUREG-1434 does not include the use of cross references. These unneeded cross references are deleted.~~

C.44 The TS need not require an administrative letter be issued to station personnel on an annual basis describing responsibility to the Shift Superintendent. Organizational responsibilities are adequately described by the stations internal administrative controls. Repeating the organizational responsibilities via an internal management directive only increases the administrative burden on the facility with no resulting benefit. Plant safety is not compromised by this proposed change.

C.45 This comment number is not used for this station.

C.46 The generic example is removed.

10/27/94