Enclosure 5

Edwin I. Hatch Nuclear Plant
Request to Implement an Alternative Source Term

Marked-up TS and Bases Pages

TABLE OF CONTENTS (continued)

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be 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. 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, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. 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 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131 Insert 1.1	~~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"; Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity.~~"

(continued)

INSERT 1.1

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same Committed Effective Dose Equivalent as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ 0.2 μCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 μCi/gm and ≤ 4.0 μCi/gm DOSE EQUIVALENT I-131.	------------------NOTE--------------- LCO 3.0.4.c is applicable. ---	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Reactor coolant specific activity > 4.0 μCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u>	
	B.2.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV (of a representative sample) actuates to restrict flow to within limits.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP system.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 28.0 psig.~~ Insert 1	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Deleted	
SR 3.6.1.3.12	Cycle each 18 inch excess flow isolation damper to the fully closed and fully open position.	24 months

Insert 2

Insert 1

Verify combined MSIV leakage rate for all four main steam lines is $\leq$ 100 scfh when tested at $\geq$ 28.0 psig and < 50.8 psig.

<u>OR</u>

Verify combined MSIV leakage rate for all four main steam lines is $\leq$ 144 scfh when tested at $\geq$ 50.8 psig.

Insert 2

SR 3.6.1.3.13	Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.02\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.2.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.2	Verify each drywell spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage.

TABLE OF CONTENTS (continued)

(continued)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be 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. 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, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS: a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and b. Control rod movement, provided there are no fuel assemblies in the associated core cell. 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 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131 Insert 1.1	~~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"; Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~

(continued)

INSERT 1.1

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same Committed Effective Dose Equivalent as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ 0.2 μCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 μCi/gm and ≤ 4.0 μCi/gm DOSE EQUIVALENT I-131. 2	------------------NOTE---------------- LCO 3.0.4.c is applicable. ---	
	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. 2 OR Reactor coolant specific activity > 4.0 μCi/gm DOSE EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	B.2.1 Isolate all main steam lines.	12 hours
	OR	
	B.2.2.1 Be in MODE 3.	12 hours
	AND	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV (of a representative sample) actuates to restrict flow to within limits.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP system.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.~~009~~ 02 L_a when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Insert 1 ~~Verify leakage rate through each MSIV is ≤ 100 scfh, and a combined maximum pathway leakage ≤ 250 scfh for all four main steam lines, when tested at ≥ 28.8 psig.~~ ~~However, the leakage rate acceptance criteria for the first test following discovery of leakage through an MSIV not meeting the 100 scfh limit, shall be ≤ 11.5 scfh for that MSIV.~~	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Deleted	
SR 3.6.1.3.13	Cycle each 18 inch excess flow isolation damper to the fully closed and fully open position.	24 months

Insert 1

Verify combined MSIV leakage rate for all four main steam lines is ≤ 100 scfh when tested at ≥ 28.8 psig and < 47.3 psig.

<u>OR</u>

Verify combined MSIV leakage rate for all four main steam lines is ≤ 144 scfh when tested at ≥ 47.3 psig.

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.2.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1 Restore RHR drywell spray subsystem to OPERABLE status.	7 days
B. Two RHR drywell spray subsystems inoperable.	B.1 Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.1	Verify each RHR drywell spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.2.5.2	Verify each drywell spray nozzle is unobstructed.	Following maintenance which could result in nozzle blockage.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active fuel must be adjusted for assemblies with a fuel length not 150 inches. For example, the top of the active fuel for GE13 fuel is 162.44 inches below instrument zero since the fuel length for this fuel type is 146 inches. The Core Operating Limits Report identifies fuel types and fuel lengths used in the current operating cycle.

SAFETY LIMITS

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 in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

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. 3).

2.2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive ~~releases~~ doses in excess of 10 CFR ~~100,~~ ~~"Reactor Site Criteria,"~~ 50.67 limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs 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.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, and the Safety Review Board (SRB) shall be notified within 24 hours. The 24 hour period provides time for plant operators and

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.3 (continued)

staff to take the immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

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

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 Fuels" (revision specified in the COLR).
3. 10 CFR 50.72.
4. 10 CFR ~~100~~ 50.67.
5. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 *Reactor Coolant System (RCS) Pressure SL*

BASES

BACKGROUND

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. Per 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).

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 may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

doses

50.67

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive ~~releases~~ from exceeding the limits specified in 10 CFR ~~100, "Reactor Site Criteria"~~ (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Addenda through the Winter of 1966 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, 1967 Edition, including Addenda A, C, and D (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1150 psig for suction piping and 1325 psig for discharge piping. The most limiting of these two allowances is the 110% of the reactor vessel design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

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. 7).

2.2.2

50.67

doses

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.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, and the SRB shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility, and the SRB.

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 and GDC 15.

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

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. 10 CFR ~~100~~. 50.67

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.

6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition, Addenda A, C, and D.

7. 10 CFR 50.72.

8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

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 to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," (revision specified in the COLR).

2. Letter from T. A. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1988.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR ~~100.11~~. 50.67

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

7. ASME, Boiler and Pressure Vessel Code.

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

9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

s

Insert A

BASES

BACKGROUND

The SLC System ~~is designed to~~ provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a sodium pentaborate solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 800 ppm of natural boron equivalent, in the reactor coolant at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The Region A volume versus concentration limits in Figure 3.1.7-1 and the Region A temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with high water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

Insert B

(continued)

Insert A

Additionally, the SLC system provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a Design Basis Accident (DBA) LOCA involving fuel damage. Maintaining the suppression pool pH at or above 7.0 will preclude the re-evolution of iodine from the suppression pool water following a DBA LOCA.

Insert B

The SLC system is also used to control suppression pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. The sodium pentaborate is then transported to the suppression pool and mixed by ECCS flow recirculation through the reactor, out of the break, and into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA radiological analysis. This quantity is maintained in the storage tank as specified in the Technical Specifications.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLC System satisfies Criterion 4 of the NRC Policy Statement (Ref. 3).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

and provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, with the mode switch in shutdown, control rod block prevents withdrawal of control rods. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM [LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"] ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

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If the sodium pentaborate solution concentration is not within the 10 CFR 50.62 limits (not within Region A of Figure 3.1.7-1 or 3.1.7-2), but greater than original licensing basis limits (within Region B of Figure 3.1.7-1 or 3.1.7-2), the solution must be restored to within Region A limits in 72 hours. It should be noted that the lowest acceptable concentration in Region B is 5%. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable, since the SLC subsystems are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits. The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed

(continued)

BASES

ACTIONS

A.1 (continued)

for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned to OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

and provide adequate buffering agent to the suppression pool.

s

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function~~.~~ However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System ~~shutdown~~ capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a ~~Design Basis Accident (~~DBA~~)~~ or severe transient occurring ~~concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.~~ The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

requiring SLC injection.

BASES

ACTIONS

B.1 (continued)

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring ~~concurrent with the failure of the control rods to shut down the reactor.~~ requiring SLC injection.

D.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained (within Region A limits of Figures 3.1.7-1 and 3.7.1-2). Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.5 (continued)

to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Additionally, the minimum pump flow rate requirement ensures that adequate buffering agent will reach the suppression pool to maintain pH at or above 7.0 post-LOCA.

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1232 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the sodium pentaborate solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency of SR 3.1.7.8 is based on a review of the surveillance test history and Reference 4.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR ~~100~~ (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram [LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"] is initiated if the SDV water level in

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 3.4.

2. 10 CFR ~~100~~ 50.67.

3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

5. NRC Safety Evaluation Report for Amendment 232.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials into the reactor coolant. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in Reference 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection systems) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20~~,~~ 50~~, and 100~~. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations include:

and

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet and cladding.

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit and certain other fuel design limits described in reference 1 are not exceeded during

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

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." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

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 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 assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four 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 MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ limits.

50.67

This Function isolates the Group 1 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch 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 valves.

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) (Ref. 2). 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 and offsite doses do not exceed the 10 CFR ~~100~~ 50.67 limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e., 1.f. Area Temperature - High

Area temperature 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 FSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Area temperature signals are initiated from RTDs (for the Main Steam Tunnel Temperature - High Function) or temperature switches (for the Turbine Building Area Temperature - High Function) located in the area being monitored. While 16 channels of Main Steam Tunnel Temperature - High Function are available, only 12 channels (6 per trip system) are required to be OPERABLE. This will ensure that no single instrument failure can preclude the isolation function, assuming a line break on any line (the instruments assigned to monitor one line can still detect a leak on another line due to their close proximity to one another and the small confines of the area). While 64 channels of Turbine Building Area Temperature - High Function are available, only 32 channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each channel has one temperature element. The 32 channel requirement is further divided up, as noted in footnote (b), into 16 channels per trip system with 8 per trip string. Each trip string shall have 2 channels per main steam line, with no more than 40 feet separating any two OPERABLE channels. In addition, no unmonitored area should exceed 40 feet in length.

The ambient temperature monitoring Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

These Functions isolate the Group 1 valves.

2. Primary Containment Isolation

50.67

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

Low RPV water level indicates that 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 3 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ are not exceeded. The Reactor Vessel Water Level - Low, Level 3

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level - Low, Level 3 (continued)

Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

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 of 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.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 6, 10, and 11 valves.

2.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

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.

This Function isolates the Group 2, 10, and 11 valves.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor 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 that 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~~ 50.67 (Ref. 1).

This LCO contains the iodine specific activity limit. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable level is intended to limit ~~the 2 hour radiation~~ offsite doses ~~to an individual at the site boundary~~ to ~~be well within~~ a small fraction of the10 CFR ~~100~~ 50.67 limits.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in References 2 and 3. 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.

This MSLB release forms the basis for determining offsite doses (Refs. 2 and 3). The limits on the specific activity of the primary coolant ensure that offsite doses, ~~the 2 hour thyroid and whole body doses at the site boundary,~~ resulting from an MSLB outside containment during steady state operation, will be ~~well within~~ a small fraction of the dose guidelines of 10 CFR ~~100~~ 50.67.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Ref. 4).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. 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 ~~well within~~ a small fraction of the 10 CFR ~~100~~ 50.67 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, 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~~ 2.0 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 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.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

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~~ 2.0 μCi/gm, it must be determined at least once 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 ~~not well within~~ more than a small fraction of the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident. Alternatively, the plant can be placed in MODE 3 within 12 hours and in 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.

The Completion Time of once every 4 hours is 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 placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

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.

(continued)

BASES (continued)

50.67

REFERENCES

1. 10 CFR ~~100.11~~.

2. FSAR, Section 14.4.5.

3. NEDE-24011-P-A-9-US, "GE Standard Application for Reactor Fuel," Supplement for United States, September 1988.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.5 (continued)

closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that each valve will isolate in a time period less than or equal to that listed in the FSAR and that no degradation affecting valve closure since the performance of the last Surveillance has occurred. (EFCVs are not required to be tested because they have no specified time limit). The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR ~~100~~ 50.67 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

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 testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. The 24 month Frequency is based on a review of the surveillance test history and Reference 8.

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) (of a representative sample) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. (The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years

(continued)

leakage rate for all four main steam lines

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.10

Combined

The analyses in References 1 and 3 are based on leakage that is less than the specified leakage rate. ~~Leakage through each~~ MSIV must be ≤ ~~11.5~~ ≤ 100 scfh when tested at ≥ 28.0 psig. and < 50.8 psig; or combined MSIV leakage rate for all four main steam lines must be ≤ 144 scfh when tested at ≥ 50.8 psig.

The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 6).

SR 3.6.1.3.11

Deleted

SR 3.6.1.3.12

This SR provides assurance that the excess flow isolation dampers can close following an isolation signal. The 24 month Frequency is based on a review of the surveillance test history and Reference 8.

Insert SR 3.6.1.3.13 here

REFERENCES

1. ~~FSAR, Section 14.4~~ Unit 2 FSAR, Section 15.3.

2. Technical Requirements Manual, Table T7.0-1.

3. FSAR, Section 5.2.

4. 10 CFR 50, Appendix J, Option B.

5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

6. Primary Containment Leakage Rate Testing Program.

7. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation."

8. NRC Safety Evaluation Report for Amendment 232.

Insert 3.6.1.3.13

SR 3.6.1.3.13

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 that form the basis of the FSAR (Ref. 1) are met. The secondary containment bypass leakage paths are: 1) main steam condensate drain, penetration 8; 2) reactor water cleanup, penetration 14; 3) equipment drain sump discharge, penetration 18; 4) floor drain sump discharge, penetration 19; 5) HPCI steam line condensate to main condenser, penetration 11; and 6) RCIC steam line condensate to main condenser, penetration 10. 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. The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 6).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

BASES

BACKGROUND

The Drywell Spray is a mode of the RHR system which may be initiated under post accident conditions to reduce the temperature and pressure of the primary containment atmosphere. Each of the two RHR subsystems consists of two pumps, one heat exchanger, containment spray valves, and a spray header in the drywell. RHR drywell spray is a manually initiated function which can only be placed in service if adequate core cooling is assured. A physical interlock prevents opening the spray valves unless reactor water level is above two thirds core height. However, under certain conditions as delineated by the emergency operating procedures, this interlock may be bypassed.

Water is pumped from the suppression pool and through the RHR heat exchangers, after which it is diverted to the spray headers in the drywell. The spray then effects a temperature and pressure reduction through the combined effects of evaporative and convective cooling, depending on the drywell atmosphere. If the atmosphere is superheated, a rapid evaporative cooling process will ensue. If the environment in the drywell is saturated, temperature and pressure will be reduced via a convective cooling process.

The drywell spray is also operated post-LOCA to wash, or scrub, inorganic iodines and particulates from the drywell atmosphere into the suppression pool.

At Plant Hatch, the drywell spray is credited post-LOCA for both the scrubbing function as well as the temperature and pressure reduction effects. The drywell spray is not credited in determining the post-LOCA peak primary containment internal pressure; however, the Hatch radiological dose analysis does take credit for the drywell spray temperature and pressure reduction over time in reducing the post-LOCA primary containment leakage and main steam isolation valve leakage.

RHR Service Water (RHRSW), circulating through the tube side of the heat exchangers, supports the drywell spray temperature and pressure reduction function by exchanging heat with the suppression pool water and discharging the heat to the external heat sink.

The drywell spray mode of RHR is described in the FSAR, Reference 1.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RHR drywell spray is credited post-LOCA for scrubbing inorganic iodines and particulates from the primary containment atmosphere. This function reduces the amount of airborne activity available for leakage from the primary containment. The RHR drywell spray also reduces the temperature and pressure in the drywell over time, thereby reducing the post-LOCA primary containment and main steam isolation valve leakage to within the assumptions of the Hatch radiological dose analysis. The RHR drywell spray system is not required to maintain the primary containment peak post-LOCA pressure within design limits.

Reference 2 contains the results of analyses used to predict the effects of drywell spray on the post accident primary containment atmosphere, as well as the primary containment leak rate analysis.

The RHR drywell spray system satisfies criterion 3 of the NRC Policy Statement (Reference 3).

LCO

In the event of a LOCA, a minimum of one RHR drywell spray subsystem using one RHR pump is required to adequately scrub the inorganic iodines and particulates from the primary containment atmosphere. One RHR drywell spray system using one RHR pump is also adequate to reduce the primary containment temperature and pressure to maintain the primary containment and main steam isolation valve post-accident leakage rates within the limits assumed in the Hatch radiological dose analysis.

To ensure these requirements are met, two RHR drywell spray subsystems must be OPERABLE with power supplies from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single failure.

An RHR drywell spray subsystem is considered OPERABLE when one of the two pumps in the subsystem, the heat exchanger, associated piping, valves, instrumentation, and controls are OPERABLE.

Each RHR drywell spray subsystem is supported by an independent subsystem of the RHRSW system. Specifically, two RHRSW pumps and an OPERABLE flow path are required to provide the necessary heat transfer from the heat exchanger and thereby support each drywell spray subsystem.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause the pressurization of, and the release of fission products into, the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining RHR drywell spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one drywell spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR drywell spray subsystem is adequate to perform the primary containment fission product scrubbing and temperature and pressure reduction functions.

However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in the loss of the scrubbing and temperature and pressure reduction capabilities of the RHR drywell spray system. The 7 day Completion Time was chosen because of the capability of the redundant and OPERABLE RHR drywell spray subsystem and the low probability of a DBA occurring during this period.

B.1

With both RHR drywell spray subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. In this Condition, there is a substantial loss of the fission product scrubbing and temperature and pressure reduction functions of the RHR drywell spray system. The 8 hour Completion Time is based on the low probability of a DBA during this period.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and 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 without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR drywell spray flow path provides assurance that the proper flow paths will exist for system 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 non-accident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR drywell spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is manually initiated. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.5.2

This surveillance is performed following maintenance which could result in nozzle blockage to verify that the spray nozzles are not obstructed and that flow will be provided when required. The frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR Section 4.8.
2. Unit 2 FSAR, Section 15.3.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Control Room Environmental Control (MCREC) System

BASES

BACKGROUND

The MCREC System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of MCREC System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of recirculated air and outside supply air. Each subsystem consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a booster fan, and the associated ductwork and dampers. Additionally, one air handling unit (AHU) fan is required for each subsystem to assist in the pressurization function. AHU fans are also addressed as part of LCO 3.7.5, "Control Room Air Conditioning (AC) System." Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The MCREC System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the MCREC System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems. Outside air is taken in at the normal ventilation intake and is mixed with the recirculated air before being passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles and gaseous iodines.

The MCREC System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding ~~5 rem whole body dose or its equivalent to any part of the body~~ the dose limits of 10 CFR 50.67. A single MCREC subsystem will pressurize the control room to ≥ 0.1 inches water gauge to prevent infiltration of air from surrounding buildings. MCREC System operation in maintaining control room habitability is discussed in the Unit 2 FSAR, Sections 6.4 and 9.4.1, (Refs. 1 and 2, respectively).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the MCREC System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Section 5.2 and Chapter 14 (Refs. 3 and 4, respectively). The pressurization mode of the MCREC System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the Unit 2 FSAR, Section 6.4.1.2.2 (Ref. 5). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 6. No single active or passive failure will cause the loss of outside air or recirculated air from the control room.

The MCREC System satisfies Criterion 3 of the NRC Policy Statement (Ref. 7).

LCO

Two redundant subsystems of the MCREC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding ~~a dose of 5 rem to~~ the 10 CFR 50.67 dose limits (Ref. 10) for the control room operators in the event of a DBA.

The MCREC System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Filter booster fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions;

c. Associated ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained;

d. One AHU fan is OPERABLE, and either operating or having its control switch in "Standby" with OPERABLE automatic start capability; and

e. Associated AHU cooling coils, water cooled condensing units, refrigerant compressors, and associated instrumentation and controls to ensure loop seal is maintained.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.4 (continued)

pressure at a flow rate of ≤ 2750 cfm through the control room in the pressurization mode. This SR ensures the total flow rate meets the design analysis value of 2500 cfm ± 10% and ensures the outside air flow rate is ≤ 400 cfm. The 24 month Frequency, on a STAGGERED TEST BASIS, is based on a review of the surveillance test history and Reference 9.

REFERENCES

1. Unit 2 FSAR, Section 6.4.
2. Unit 2 FSAR, Section 9.4.1.
3. FSAR, Section 5.2.
4. FSAR, Chapter 14.
5. Unit 2 FSAR, Section 6.4.1.2.2.
6. Unit 2 FSAR, Table 15.1-28.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. Technical Requirements Manual, Table T2.1-1.
9. NRC Safety Evaluation Report for Amendment 232.
10. 10 CFR 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section 9.4 and Appendix E (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 µCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 µCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2804 MWt.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

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-85m, 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.

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

1. FSAR, Section 9.4 and Appendix E.
2. 10 CFR ~~100~~. 50.67
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 10.3 (Ref. 1). The assumptions of the fuel handling accident in the spent fuel storage pool are found in Reference 2.

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident; the point from which the water level is measured is shown in Figure B 3.5.2-1. A fuel handling accident in the spent fuel storage pool was evaluated (Ref. 2) and ensured that the radiological consequences ~~(calculated whole body and thyroid doses at the exclusion area and low population zone boundaries~~) were well ~~below~~ the ~~guideline doses of~~ 10 CFR ~~100~~ (Ref. 3) and met the exposure guidelines of ~~NUREG-0800 (Ref. 4)~~. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ (Ref. 5).

dose

within

Regulatory Guide 1.183 (Ref. 5)

1.183

50.67 limits

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the spent fuel storage pool racks (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is 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 a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. FSAR, Section 10.3.
2. ~~NRC Safety Evaluation Report related to Unit 1 Amendment 172 and Unit 2 Amendment 112, August 28, 1991.~~ Unit 2 FSAR, Section 15.3.
3. 10 CFR ~~100~~ 50.67.
4. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~ Deleted.

(continued)

BASES

REFERENCES (continued)	5.	Regulatory Guide ~~1.25, March 1972~~. 1.183, July 2000.
	6.	NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV. The point from which the water level is measured is shown in Figure B.5.2-1. During refueling, this maintains a sufficient water level in the reactor vessel cavity. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well ~~below~~ within the ~~guideline doses of~~ 10 CFR ~~100~~ 50.67 limits, as provided by the guidance of Reference ~~3~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). ~~A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 4). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

RPV water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

A minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV is required to ensure that the

(continued)

BASES

LCO (continued)

radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference 3. The point from which the water level is measured is shown in Figure B 3.5.2-1.

1

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.8, "Spent Fuel Storage Pool Water Level."

ACTIONS

A.1

If the water level is < 23 ft above the top of the irradiated fuel assemblies seated within the RPV, all operations involving movement of fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000.

2. ~~FSAR, Section 14.4.4.~~ Unit 2 FSAR, Section 15.3.

3. ~~NUREG-0800, Section 15.7.4.~~ Deleted.

4. 10 CFR ~~100.11~~ 50.67.

5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active fuel must be adjusted for assemblies with a fuel length not 150 inches. For example, the top of the active fuel for GE13 fuel is 162.44 inches below instrument zero since the fuel length for this fuel type is 146 inches. The Core Operating Limits Report identifies fuel types and fuel lengths used in the current operating cycle.

SAFETY LIMITS

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 in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

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. 3).

2.2.2

doses 50.67

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). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs 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.

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, and the Safety Review Board (SRB) shall be notified within 24 hours. The 24 hour period provides time for plant operators and

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.3 (continued)

staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

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

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 Fuels," (revision specified in the COLR).
3. 10 CFR 50.72.
4. 10 CFR ~~100~~ 50.67.
5. 10 CFR 50.73.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

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. Per 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).

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 may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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~~ doses from exceeding the limits specified in 10 CFR ~~100, "Reactor Site Criteria"~~ 50.67 (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure - High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1968 Edition, including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	Addenda through the Summer of 1970 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1980 Edition, including addenda through Winter 1981 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping and 1450 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.
SAFETY LIMITS	The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping and 1450 psig for discharge piping. The most limiting of these two allowances is the 110% of the reactor vessel and recirculation suction piping design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.
APPLICABILITY	SL 2.1.2 applies in all MODES.
SAFETY LIMIT VIOLATIONS	<u>2.2.1</u> If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7). <u>2.2.2</u> 50.67 doses 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.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If any SL is violated, the senior management of the nuclear plant and the utility, and the SRB shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant and the utility, and the SRB.

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 and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. 10 CFR ~~100~~ 50.67.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition, Addenda Summer of 1970.
6. ASME, Boiler and Pressure Vessel Code, Section III, 1980 Edition, Addenda Winter of 1981.
7. 10 CFR 50.72.
8. 10 CFR 50.73.

BASES

ACTIONS

B.1 and B.2 (continued)

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 to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," (revision specified in the COLR).
2. Letter from T. A. Pickens (BWROG) to G. C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," BWROG-8644, August 15, 1988.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
5. 10 CFR ~~100.11~~ 50.67.
6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
7. ASME, Boiler and Pressure Vessel Code.
8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

Insert A

s

BASES

BACKGROUND

The SLC System ~~is designed to~~ provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a sodium pentaborate solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 800 ppm of natural boron equivalent, in the reactor coolant at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The Region A volume versus concentration limits in Figure 3.1.7-1 and the Region A temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with high water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

Insert B

(continued)

Insert A

Additionally, the SLC system provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a Design Basis Accident (DBA) LOCA involving fuel damage. Maintaining the suppression pool pH at or above 7.0 will preclude the re-evolution of iodine from the suppression pool water following a DBA LOCA.

Insert B

The SLC system is also used to control suppression pool pH in the event of a DBA LOCA by injecting sodium pentaborate into the reactor vessel. The sodium pentaborate is then transported to the suppression pool and mixed by ECCS flow recirculation through the reactor, out of the break, and into the suppression chamber. The amount of sodium pentaborate solution that must be available for injection following a DBA LOCA is determined as part of the DBA LOCA radiological analysis. This quantity is maintained in the storage tank as specified in the Technical Specifications.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SLC System satisfies Criterion 4 of the NRC Policy Statement (Ref. 3).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

and provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a DBA LOCA involving fuel damage

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, with the mode switch in shutdown, control rod block prevents withdrawal of control rods. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM [LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"] ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If the sodium pentaborate solution concentration is not within the 10 CFR 50.62 limits (not within Region A of Figure 3.1.7-1 or 3.1.7-2), but greater than original licensing basis limits (within Region B of Figure 3.1.7-1 or 3.1.7-2), the solution must be restored to within Region A limits in 72 hours. It should be noted that the lowest acceptable concentration in Region is 5%. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable, since the SLC subsystems are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits. The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed

S

(continued)

BASES

ACTIONS

A.1 (continued)

for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned to OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

s

and provide adequate buffering agent to the suppression pool.

B.1

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System ~~shutdown~~ capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a ~~Design Basis Accident (~~DBA~~)~~ or severe transient occurring ~~concurrent with the failure of the Control Rod Drive (CRD) System to shut down the plant.~~ The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

requiring SLC injection.

(continued)

BASES

ACTIONS

B.1 (continued)

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring ~~concurrent with the failure of the control rods to shut down the reactor.~~ requiring SLC injection.

D.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to 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

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained (within Region A limits of Figures 3.1.7-1 and 3.1.7-2). Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.5 (continued)

to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Additionally, the minimum pump flow rate requirement ensures that adequate buffering agent will reach the suppression pool to maintain pH at or above 7.0 post-LOCA.

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1232 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the sodium pentaborate solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency of SR 3.1.7.8 is based on a review of the surveillance test history and Reference 4.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 4.2.3.2.2.3.
2. 10 CFR ~~100~~ 50.67.
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
5. NRC Safety Evaluation Report for Amendment 174.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials into the reactor coolant. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 2.

and

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in Reference 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection systems) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50~~, and 100~~. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations include:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet and cladding.

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit and certain other fuel design limits described in reference 1 are not exceeded during

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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." Functions that have different Applicabilities are discussed below in the individual Functions discussion.

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 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 assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four 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 MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ limits.

50.67

This Function isolates the Group 1 valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch 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 valves.

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) (Ref. 2). 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 and offsite doses do not exceed the 10 CFR ~~100~~ limits.

50.67

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.e., 1.f. Area Temperature - High

Area temperature 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 FSAR, since bounding analyses are performed for large breaks, such as MSLBs.

Area temperature signals are initiated from RTDs (for the Main Steam Tunnel Temperature - High Function) or a thermocouple/temperature switch combination (for the Turbine Building Area Temperature - High Function) located in the area being monitored. While 16 channels of Main Steam Tunnel Temperature - High Function are available, only 12 channels (6 per trip system) are required to be OPERABLE. This will ensure that no single instrument failure can preclude the isolation function, assuming a line break on any line (the instruments assigned to monitor one line can still detect a leak on another line due to their close proximity to one another and the small confines of the area). While 64 channels of Turbine Building Area Temperature - High Function are available, only 32 channels are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Each channel has one temperature element. The 32 channel requirement is further divided up, as noted in footnote (b), into 16 channels per trip system with 8 per trip string. Each trip string shall have 2 channels per main steam line, with no more than 40 feet separating any two OPERABLE channels. In addition, no unmonitored area should exceed 40 feet in length.

The ambient temperature monitoring Allowable Value is chosen to detect a leak equivalent to between 1% and 10% rated steam flow.

These Functions isolate the Group 1 valves.

2. Primary Containment Isolation

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

Low RPV water level indicates that 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 3 supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded. The Reactor Vessel Water Level - Low, Level 3

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Reactor Vessel Water Level - Low, Level 3 (continued)

Function associated with isolation is implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

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 of 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.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 6, 7, 10, 11, and 12 valves.

2.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR ~~100~~ 50.67 are not exceeded. The Drywell Pressure - High Function, associated with isolation of the primary containment, is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

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.

This Function isolates the Group 2, 7, 10, 11, and 12 valves.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor 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 that 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~~ 50.67 (Ref. 1).

This LCO contains the iodine specific activity limit. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable level is intended to limit ~~the 2 hour radiation dose to an individual at the site boundary~~ offsite doses to ~~be well within~~ a small fraction of the 10 CFR ~~100~~ 50.67 limits.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in References 2 and 3. 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.

This MSLB release forms the basis for determining offsite doses (Refs. 2 and 3). The limits on the specific activity of the primary coolant ensure that ~~the 2 hour thyroid and whole body doses at the site boundary,~~ offsite doses, resulting from an MSLB outside containment during steady state operation, will be ~~well within~~ a small fraction of the dose guidelines of 10 CFR ~~100~~ 50.67.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Ref. 4).

LCO

The specific iodine activity is limited to ≤ 0.2 µCi/gm DOSE EQUIVALENT I-131. 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 ~~well within~~ the 10 CFR ~~100~~ limits.

50.67

a small fraction of

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, 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

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 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.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

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~~ 2.0 µCi/gm, it must be determined at least once 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 ~~not well within~~ more than a small fraction of the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident. Alternatively, the plant can be placed in MODE 3 within 12 hours and in 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.

The Completion Time of once every 4 hours is 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 placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

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.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR ~~100.11~~ 50.67.

2. FSAR, Section 15.1.40.

3. NEDE-24011-P-A-9-US, "GE Standard Application for Reactor Fuel," Supplement for United States, September 1988.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.5 (continued)

closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that each valve will isolate in a time period less than or equal to that listed in the FSAR and that no degradation affecting valve closure since the performance of the last surveillance has occurred. (EFCVs are not required to be tested because they have no specified time limit). The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR ~~100~~ 50.67 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

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 testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. The 24 month Frequency is based on a review of the surveillance test history and Reference 9.

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) (of a representative sample) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. (The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.10

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 that form the basis of the FSAR (Ref. 3) are met. The secondary containment bypass leakage paths are: 1) main steam condensate drain, penetration 8; 2) reactor water cleanup, penetration 14; 3) equipment drain sump discharge, penetration 18; 4) floor drain sump discharge, penetration 19; 5) chemical drain sump discharge, penetration 55; 6) HPCI steam line condensate to main condenser, penetration 11; and 7) RCIC steam line condensate to main condenser, penetration 10. 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. The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 7).

leakage rate for all four main steam lines

SR 3.6.1.3.11

Combined

The analyses in References 1 and 4 are based on leakage that is less than the specified leakage rate. ~~Leakage through each~~ MSIV must be ≤ 100 scfh~~, and a combined maximum pathway leakage ≤ 250 scfh for all four main steam lines~~ when tested at ≥ 28.8 psig. ~~In addition, if any MSIV exceeds the 100 scfh limit, the as left leakage shall be ≤ 11.5 scfh for that MSIV.~~

The Frequency is required by the Primary Containment Leakage Rate Testing Program.

and < 47.3 psig; or combined MSIV leakage rate for all four main steam lines must be ≤ 144 scfh when tested at ≥ 47.3 psig.

SR 3.6.1.3.12

Deleted

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.5 Residual Heat Removal (RHR) Drywell Spray

BASES

BACKGROUND

The Drywell Spray is a mode of the RHR system which may be initiated under post accident conditions to reduce the temperature and pressure of the primary containment atmosphere. Each of the two RHR subsystems consists of two pumps, one heat exchanger, containment spray valves, and a spray header in the drywell. RHR drywell spray is a manually initiated function which can only be placed in service if adequate core cooling is assured. A physical interlock prevents opening the spray valves unless reactor water level is above two thirds core height. However, under certain conditions as delineated by the emergency operating procedures, this interlock may be bypassed.

Water is pumped from the suppression pool and through the RHR heat exchangers, after which it is diverted to the spray headers in the drywell. The spray then effects a temperature and pressure reduction through the combined effects of evaporative and convective cooling, depending on the drywell atmosphere. If the atmosphere is superheated, a rapid evaporative cooling process will ensue. If the environment in the drywell is saturated, temperature and pressure will be reduced via a convective cooling process.

The drywell spray is also operated post-LOCA to wash, or scrub, inorganic iodines and particulates from the drywell atmosphere into the suppression pool.

At Plant Hatch, the drywell spray is credited post-LOCA for both the scrubbing function as well as the temperature and pressure reduction effects. The drywell spray is not credited in determining the post-LOCA peak primary containment internal pressure; however, the Hatch radiological dose analysis does take credit for the drywell spray temperature and pressure reduction over time in reducing the post-LOCA primary containment leakage and main steam isolation valve leakage.

RHR Service Water (RHRSW), circulating through the tube side of the heat exchangers, supports the drywell spray temperature and pressure reduction function by exchanging heat with the suppression pool water and discharging the heat to the external heat sink.

The drywell spray mode of RHR is described in the FSAR, Reference 1.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RHR drywell spray is credited post-LOCA for scrubbing inorganic iodines and particulates from the primary containment atmosphere. This function reduces the amount of airborne activity available for leakage from the primary containment. The RHR drywell spray also reduces the temperature and pressure in the drywell over time, thereby reducing the post-LOCA primary containment and main steam isolation valve leakage to within the assumptions of the Hatch radiological dose analysis. The RHR drywell spray system is not required to maintain the primary containment peak post-LOCA pressure within design limits.

Reference 2 contains the results of analyses used to predict the effects of drywell spray on the post accident primary containment atmosphere, as well as the primary containment leak rate analysis.

The RHR drywell spray system satisfies criterion 3 of the NRC Policy Statement (Reference 3).

LCO

In the event of a LOCA, a minimum of one RHR drywell spray subsystem using one RHR pump is required to adequately scrub the inorganic iodines and particulates from the primary containment atmosphere. One RHR drywell spray system using one RHR pump is also adequate to reduce the primary containment temperature and pressure to maintain the primary containment and main steam isolation valve post-accident leakage rates within the limits assumed in the Hatch radiological dose analysis.

To ensure these requirements are met, two RHR drywell spray subsystems must be OPERABLE with power supplies from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single failure.

An RHR drywell spray subsystem is considered OPERABLE when one of the two pumps in the subsystem, the heat exchanger, associated piping, valves, instrumentation, and controls are OPERABLE.

Each RHR drywell spray subsystem is supported by an independent subsystem of the RHRSW system. Specifically, two RHRSW pumps and an OPERABLE flow path are required to provide the necessary heat transfer from the heat exchanger and thereby support each drywell spray subsystem.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause the pressurization of, and the release of fission products into, the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining RHR drywell spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one drywell spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR drywell spray subsystem is adequate to perform the primary containment fission product scrubbing and temperature and pressure reduction functions.

However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in the loss of the scrubbing and temperature and pressure reduction capabilities of the RHR drywell spray system. The 7 day Completion Time was chosen because of the capability of the redundant and OPERABLE RHR drywell spray subsystem and the low probability of a DBA occurring during this period.

B.1

With both RHR drywell spray subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. In this Condition, there is a substantial loss of the fission product scrubbing and temperature and pressure reduction functions of the RHR drywell spray system. The 8 hour Completion Time is based on the low probability of a DBA during this period.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and 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 without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR drywell spray flow path provides assurance that the proper flow paths will exist for system 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 non-accident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR drywell spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is manually initiated. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.5.2

This surveillance is performed following maintenance which could result in nozzle blockage to verify that the spray nozzles are not obstructed and that flow will be provided when required. The frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR Section 5.5.7.

2. Unit 2 FSAR, Section 15.3.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Control Room Environmental Control (MCREC) System

BASES

BACKGROUND

The MCREC System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of MCREC System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of recirculated air and outside supply air. Each subsystem consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a booster fan, and the associated ductwork and dampers. Additionally, one air handling unit (AHU) fan is required for each subsystem to assist in the pressurization function. AHU fans are also addressed as part of LCO 3.7.5, "Control Room Air Conditioning (AC) System." Prefilters and HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The MCREC System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the MCREC System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room, and a part of the recirculated air is routed through either of the two filter subsystems. Outside air is taken in at the normal ventilation intake and is mixed with the recirculated air before being passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles and gaseous iodines.

the dose limits of 10 CFR 50.67.

The MCREC System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding ~~5 rem whole body dose or its equivalent to any part of the body.~~ A single MCREC subsystem will pressurize the control room to ≥ 0.1 inches water gauge to prevent infiltration of air from surrounding buildings. MCREC System operation in maintaining control room habitability is discussed in the FSAR, Sections 6.4 and 9.4.1, (Refs. 1 and 2, respectively).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the MCREC System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters 6 and 15 (Refs. 3 and 4, respectively). The pressurization mode of the MCREC System is assumed to operate following a loss of coolant accident, fuel handling accident, main steam line break, and control rod drop accident, as discussed in the FSAR, Section 6.4.1.2.2 (Ref. 5). The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 6. No single active or passive failure will cause the loss of outside air or recirculated air from the control room.

The MCREC System satisfies Criterion 3 of the NRC Policy Statement (Ref. 7).

LCO

Two redundant subsystems of the MCREC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding ~~a dose of 5 rem to~~ the 10 CFR 50.67 dose limits (Ref. 10) for the control room operators in the event of a DBA.

The MCREC System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Filter booster fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow and are capable of performing their filtration functions;

c. Associated ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained;

d. One AHU fan is OPERABLE, and either operating or having its control switch in "Standby" with OPERABLE automatic start capability; and

e. Associated AHU cooling coils, water cooled condensing units, refrigerant compressors, and associated instrumentation and controls to ensure loop seal is maintained.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.4 (continued)

pressure at a flow rate of ≤ 2750 cfm through the control room in the pressurization mode. This SR ensures the total flow rate meets the design analysis value of 2500 cfm ± 10% and ensures the outside air flow rate is ≤ 400 cfm. The 24 month Frequency, on a STAGGERED TEST BASIS, is based on a review of the surveillance test history and Reference 9.

REFERENCES

1. FSAR, Section 6.4.
2. FSAR, Section 9.4.1.
3. FSAR, Chapter 6.
4. FSAR, Chapter 15.
5. FSAR, Section 6.4.1.2.2.
6. FSAR, Table 15.1-28.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. Technical Requirements Manual, Table T2.1-1.
9. NRC Safety Evaluation Report for Amendment 174.
10. 10 CFR 50.67.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Sections 11.3 and 15.1.35 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR ~~100~~ (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

50.67

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2804 MWt.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

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-85m, 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.

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

1. FSAR, Sections 11.3 and 15.1.35.
2. 10 CFR ~~100~~ 50.67.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident in the spent fuel storage pool are found in Reference 2.

APPLICABLE SAFETY ANALYSES

dose

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident; the point from which the water level is measured is shown in Figure B 3.5.2-1. A fuel handling accident in the spent fuel storage pool was evaluated (Ref. 2) and ensured that the radiological consequences ~~(calculated whole body and thyroid doses at the exclusion area and low population zone boundaries)~~ were well ~~below~~ within the ~~guideline doses of~~ 10 CFR ~~100~~ 50.67 limits (Ref. 3) and met the exposure guidelines of ~~NUREG-0800 (Ref. 4)~~ Regulatory Guide 1.183 (Ref. 5). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the spent fuel storage pool racks (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel movement within the spent fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is 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 a sufficient reason to require a reactor shutdown.

When the initial conditions for an accident cannot be met, action must be taken to preclude the accident from occurring. If the spent fuel storage pool level is less than required, the movement of irradiated fuel assemblies in the spent fuel storage pool is suspended immediately. Suspension of this activity shall not preclude completion of movement of an irradiated fuel assembly to a safe position. This effectively precludes a spent fuel handling accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR verifies that sufficient water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is acceptable, based on operating experience, considering that the water volume in the pool is normally stable, and all water level changes are controlled by unit procedures.

REFERENCES

1. FSAR, Section 9.1.2. FSAR, Section 15.3.

2. ~~NRC Safety Evaluation Report related to Unit 1 Amendment 172 and Unit 2 Amendment 112, August 28, 1991.~~

3. 10 CFR ~~100~~. 50.67

4. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~

Deleted.

(continued)

BASES

REFERENCES (continued)	5.	Regulatory Guide ~~1.25, March 1972~~. 1.183, July 2000.
	6.	NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV. The point from which the water level is measured is shown in Figure B 3.5.2-1. During refueling, this maintains a sufficient water level in the reactor vessel cavity. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well ~~below~~ the ~~guideline doses of~~ 10 CFR ~~100~~, as provided by the guidance of Reference ~~3~~.

50.67 limits

within

1

APPLICABLE SAFETY ANALYSES

1.183

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide ~~1.25~~ (Ref. 1). ~~A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 4). The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

RPV water level satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

A minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV is required to ensure that the

(continued)

BASES

LCO (continued)

radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference $\boxed{3}$. The point from which the water level is measured is shown in Figure B 3.5.2-1.

1

APPLICABILITY

LCO 3.9.6 is applicable when moving fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.8, "Spent Fuel Storage Pool Water Level."

ACTIONS

A.1

If the water level is < 23 ft above the top of the irradiated fuel assemblies seated within the RPV, all operations involving movement of fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972.~~ 1.183, July 2000.

2. FSAR, Section ~~15.1.41~~ 15.3.

3. ~~NUREG-0800, Section 15.7.4.~~ Deleted.

4. 10 CFR ~~100.11~~ 50.67.

5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.