

RS-01-106

May 29, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: Submittal of Inadvertently Omitted Improved Technical Specifications Bases Pages

- References:
- 1) Letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to implement Improved Standard Technical Specifications," dated March 3, 2000
 - 2) Letter from R.M. Krich (Exelon Generation Company, LLC) to U.S. NRC, "Certification of Technical Specifications and Bases Supporting the Implementation of Improved Standard Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2

In Reference 1, in accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Commonwealth Edison (ComEd) Company, now Exelon Generation Company (EGC), LLC, proposed to amend Appendix A, Technical Specifications (TS) of Facility Operating License Nos. DPR-19, DPR-25, NPF-11, NPF-18, DPR-29 and DPR-30 for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed changes revise the Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, current Technical Specifications (CTS) to a format and content consistent with NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 4," and NUREG-1434, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 6," as applicable.

In Reference 2, EGC provided final copies of the ITS for DNPS. Several of the Dresden Nuclear Power Station Bases pages were inadvertently omitted in Reference 2.

ADD

May 29, 2001
U.S. Nuclear Regulatory Commission
Page 2

The attachment to this letter provides the NRC with the missing pages. Should you have any questions concerning this transmittal, please contact Mr. Patrick Simpson at (630) 663-6698.

Respectfully,

A handwritten signature in black ink, appearing to read "R. M. Krich". The signature is fluid and cursive, with the first letters of each word being capitalized and prominent.

R. M. Krich
Director-Licensing
Mid-West Regional Operating Group

Attachments: Missing Pages from the Dresden Nuclear Power Station Bases

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - LaSalle County Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

bcc: NRC Project Manager, NRR - Dresden Nuclear Power Station, Units 2 and 3
w/o Attachments
NRC Project Manager, NRR - LaSalle County Station
w/o Attachments
NRC Project Manager, NRR - Quad Cities Nuclear Power Station
Site Vice President - Dresden Nuclear Power Station
w/o Attachments
Site Vice President - LaSalle County Station
w/o Attachments
Site Vice President - Quad Cities Nuclear Power Station
w/o Attachments
Vice President – Licensing and Regulatory Affairs
w/o Attachments
Nicholas Reynolds - Winston and Strawn
w/o Attachments
Manager, Licensing - Dresden and Quad Cities Nuclear Power Stations
w/o Attachments
Manager, Licensing - LaSalle County and Clinton Power Stations
w/o Attachments
Regulatory Assurance Manager - Dresden Nuclear Power Station
w/o Attachments
Regulatory Assurance Manager - LaSalle County Station
w/o Attachments
Regulatory Assurance Manager - Quad Cities Nuclear Power Station
w/o Attachments
Exelon Document Control Desk - Licensing (Hard Copy)
Exelon Document Control Desk - Licensing (Electronic Copy)

ATTACHMENT

Missing Pages from Dresden Nuclear Power Station Bases

The following pages are provided following this cover page.

Page Number

B 3.6.2.1-4
B 3.6.2.1-5
B 3.6.2.2-1
B 3.6.2.2-2
B 3.6.2.2-3
B 3.6.2.3-1
B 3.6.2.3-2
B 3.6.2.3-3
B 3.6.2.3-4
B 3.6.2.4-1
B 3.6.2.4-2
B 3.6.2.4-3
B 3.6.2.4-4
B 3.6.2.5-1
B 3.6.2.5-2
B 3.6.2.5-3
B 3.6.3.1-1
B 3.6.3.1-2
B 3.6.3.1-3
B 3.6.4.1-1
B 3.6.4.1-2
B 3.6.4.1-3
B 3.6.4.1-4
B 3.6.4.1-5
B 3.6.4.1-6
B 3.6.4.2-1
B 3.6.4.2-2
B 3.6.4.2-3
B 3.6.4.2-4
B 3.6.4.2-5
B 3.6.4.2-6
B 3.6.4.2-7
B 3.6.4.3-1
B 3.6.4.3-2
B 3.6.4.3-3
B 3.6.4.3-4
B 3.6.4.3-5
B 3.6.4.3-6
B 3.6.4.3-7
B 3.7.1-1
B 3.7.1-2
B 3.7.1-3
B 3.7.1-4
B 3.7.1-5
B 3.7.1-6

BASES

ACTIONS
(continued)

C.1

Suppression pool average temperature is allowed to be $> 95^{\circ}\text{F}$ with THERMAL POWER $> 1\%$ RTP, and when testing that adds heat to the suppression pool is being performed. However, if temperature is $> 105^{\circ}\text{F}$, all testing must be immediately suspended to preserve the heat absorption capability of the suppression pool. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

D.1 and D.2

Suppression pool average temperature $> 110^{\circ}\text{F}$ requires that the reactor be shut down immediately. This is accomplished by placing the reactor mode switch in the shutdown position. Further cooldown to MODE 4 within 36 hours is required at normal cooldown rates (provided pool temperature remains $\leq 120^{\circ}\text{F}$). Additionally, when suppression pool temperature is $> 110^{\circ}\text{F}$, increased monitoring of pool temperature is required to ensure that it remains $\leq 120^{\circ}\text{F}$. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained at $\leq 120^{\circ}\text{F}$, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < 150 psig within 12 hours, and the plant must be brought to at least MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

Continued addition of heat to the suppression pool with suppression pool temperature > 120°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. UFSAR, Section 6.2.
 2. NEDC-22170, Dresden 2 and 3 Nuclear Generating Plant Suppression Pool Temperature Response, July 1982.
 3. NUREG-0783.
 4. Dresden Nuclear Power Station Units 2 and 3 Plant Unique Analysis Report, COM-02-041, May 1983.
-

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from relief valve discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from the steam exhaust line in the turbine driven High Pressure Coolant Injection (HPCI) System and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 116,300 ft³ at the low water level limit of 14 ft 6.5 inches and 119,800 ft³ at the high water level limit of 14 ft 10.5 inches.

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI turbine exhaust line. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

If the suppression pool water level is too high, it could result in excessive clearing loads from relief valve discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to relief valve discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

A limit that suppression pool water level be ≥ 14 ft 6.5 inches and ≤ 14 ft 10.5 inches above the bottom of the suppression chamber is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS - Shutdown."

ACTIONS

A.1

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as the downcomers are covered, HPCI turbine exhaust is covered, and relief valve quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the Suppression Pool Spray System. Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

If suppression pool water level cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. The 24 hour Frequency has been shown to be acceptable based on operating experience. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool water level condition.

REFERENCES

1. UFSAR, Section 6.2.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each suppression pool cooling subsystem contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the containment cooling heat exchangers and returning it to the suppression pool. Containment cooling service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink.

The heat removal capability of one low pressure coolant injection (LPCI) pump in one subsystem is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open relief valve. Relief valve leakage and High Pressure Coolant Injection System testing increase suppression pool temperature more slowly. The Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The intent of the analyses is to demonstrate that the heat removal capacity of the Suppression Pool Cooling System is adequate to maintain the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

primary containment conditions within design limits. The suppression pool temperature is calculated to remain below the design limit.

The Suppression Pool Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. 1). To ensure that these requirements are met, two suppression pool cooling subsystems must be OPERABLE with power 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 active failure. A suppression pool cooling subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both a release of radioactive material to primary containment and a heatup and pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant suppression pool cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS
(continued)

B.1

With two suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and the potential avoidance of a plant shutdown transient that could result in the need for the suppression pool cooling subsystems to operate.

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 to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual and power operated valves in the suppression pool cooling mode flow path provides assurance that the proper flow path exists 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 nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the suppression pool cooling 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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.3.1 (continued)

event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each required LPCI pump develops a flow rate ≥ 5000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment peak pressure and temperature can be maintained below the design limits during a DBA (Ref. 1). The flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 6.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Suppression Pool Spray

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the Suppression Pool Spray System removes heat from the suppression chamber airspace. The suppression pool is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel (RPV) through relief valves. The heat addition to the suppression pool results in increased steam in the suppression chamber, which increases primary containment pressure. Steam blowdown from a DBA can also bypass the suppression pool and end up in the suppression chamber airspace. Some means must be provided to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. This function is provided by two redundant suppression pool spray subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each of the two suppression pool spray subsystems contains two pumps and one heat exchanger, which are manually initiated and independently controlled. The two subsystems perform the suppression pool spray function by circulating water from the suppression pool through the containment cooling heat exchangers and returning it to the suppression pool spray sparger. The sparger only accommodates a small portion of the total low pressure coolant injection (LPCI) pump flow; the remainder of the flow returns to the suppression pool through the suppression pool cooling return line or minimum flow line. Thus, both suppression pool cooling and suppression pool spray functions may be performed when the Suppression Pool Spray System is initiated. Containment cooling service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink. Either suppression pool spray subsystem is sufficient to condense the steam from small bypass leaks from the drywell to the suppression chamber airspace during the postulated DBA.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break loss of coolant accidents. The intent of the analyses is to demonstrate that the pressure reduction capacity of the Suppression Pool Spray System is adequate to maintain the primary containment conditions within design limits. The time history for primary containment pressure is calculated to demonstrate that the maximum pressure remains below the design limit.

The Suppression Pool Spray System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In the event of a DBA, a minimum of one suppression pool spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below the design limits (Ref. 1). To ensure that these requirements are met, two suppression pool spray subsystems must be OPERABLE with power 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 active failure. A suppression pool spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

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

ACTIONS

A.1

With one suppression pool spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE suppression pool spray subsystem is adequate to perform the primary containment bypass leakage mitigation function.

(continued)

BASES

ACTIONS

A.1 (continued)

However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment bypass mitigation capability. The 7 day Completion Time was chosen in light of the redundant suppression pool spray capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

With both suppression pool 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 primary containment bypass leakage mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to reduce pressure in the primary containment are available.

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 and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.4.1

Verifying the correct alignment for manual and power operated valves in the suppression pool spray mode flow path provides assurance that the proper flow path exists 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 nonaccident position provided it can be aligned to the accident position within the time assumed in the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.4.1 (continued)

accident analysis. This is acceptable since the suppression pool 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 system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.4.2

This Surveillance is performed every 10 years to verify that the spray nozzles are not obstructed and that spray flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and has been shown to be acceptable through operating experience.

REFERENCES

1. UFSAR, Section 6.2.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.5 Drywell-to-Suppression Chamber Differential Pressure

BASES

BACKGROUND

The toroidal shaped suppression chamber, which contains the suppression pool, is connected to the drywell (part of the primary containment) by eight main vent pipes. The main vent pipes exhaust into a continuous vent header, from which 96 downcomer pipes extend into the suppression pool. The pipe exit is 3.67 ft below the minimum suppression pool water level required by LCO 3.6.2.2, "Suppression Pool Water Level." During a loss of coolant accident (LOCA), the increasing drywell pressure will force the waterleg in the downcomer pipes into the suppression pool at substantial velocities as the "blowdown" phase of the event begins. The length of the waterleg has a significant effect on the resultant primary containment pressures and loads.

APPLICABLE
SAFETY ANALYSES

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. The required differential pressure results in a downcomer waterleg of 3.21 to 3.54 ft.

Initial drywell-to-suppression chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident LOCA. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid.

Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A drywell-to-suppression chamber differential pressure limit of 1.0 psid is required to ensure that the containment conditions assumed in the safety analyses are met. A

(continued)

BASES

LCO
(continued) drywell-to-suppression chamber differential pressure of < 1.0 psid corresponds to a downcomer water leg of > 3.54 ft. Failure to maintain the required differential pressure could result in excessive forces on the suppression chamber due to higher water clearing loads from downcomer vents and higher pressure buildup in the drywell.

A Note is provided to allow for periods of up to 4 hours when the LCO is not required to be met during the performance of required Surveillances that reduce the differential pressure. The 4 hour time is acceptable since the probability of a DBA LOCA occurring during this time is low.

APPLICABILITY Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The primary containment must be inert in MODE 1, since this is the condition with the highest probability for an event that could produce hydrogen. It is also the condition with the highest probability of an event that could impose large loads on the primary containment.

Inerting primary containment is an operational problem because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the unit startup and is de-inerted as soon as possible in the unit shutdown. As long as reactor power is < 15% RTP, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the first 24 hours following a startup or within the last 24 hours prior to a shutdown is low enough that these "windows," with the primary containment not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS A.1

If drywell-to-suppression chamber differential pressure is not within the limit, the conditions assumed in the safety analyses are not met and the differential pressure must be restored to within the limit within 24 hours. The 24 hour Completion Time provides sufficient time to restore

(continued)

BASES

ACTIONS

A.1 (continued)

differential pressure to within limit and takes into account the low probability of an event that would create excessive suppression chamber loads occurring during this time period.

B.1

If the differential pressure cannot be restored to within limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to $\leq 15\%$ RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.5.1

The drywell-to-suppression chamber differential pressure is regularly monitored to ensure that the required limits are satisfied. The 12 hour Frequency of this SR was developed based on operating experience relative to differential pressure variations and pressure instrument drift during applicable MODES and by assessing the proximity to the specified LCO differential pressure limit. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal pressure condition.

REFERENCES

None.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Oxygen Concentration

BASES

BACKGROUND

The primary containment is designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration < 4.0 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The capability to inert the primary containment and maintain oxygen < 4.0 v/o provides a method to mitigate events that produce hydrogen and oxygen. For example, an event that rapidly generates hydrogen from zirconium metal water reaction will result in excessive hydrogen in primary containment, but oxygen concentration will remain < 4.0 v/o and no combustion can occur. Long term generation of both hydrogen and oxygen from radiolytic decomposition of water may eventually result in a combustible mixture in primary containment. Radiolysis is the only significant reaction mechanism whereby oxygen, the limiting combustion reactant, is produced within the containment. The Technical Specification requirement to inert the primary containment and maintain oxygen < 4.0 v/o, in conjunction with the elimination of potential sources of air and oxygen (other than by radiolysis) from entering the primary containment provide assurance that the amount of oxygen that could be introduced into the containment will not cause the containment to become de-inerted within the first 30 days after an accident. This is consistent with the requirements of Generic Letter 84-09 (Ref. 1) for plants without recombiners. This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.

APPLICABLE
SAFETY ANALYSES

The Reference 2 calculations assume that the primary containment is inerted when a Design Basis Accident loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

combustible gas mixtures in the primary containment. Oxygen, which is subsequently generated by radiolytic decomposition of water, will not result in the primary containment becoming de-inerted within the first 30 days following an accident.

Primary containment oxygen concentration satisfies 10 CFR 50.36(c)(2)(ii).

LCO

The primary containment oxygen concentration is maintained < 4.0 v/o to ensure that an event that produces any amount of hydrogen and oxygen does not result in a combustible mixture inside primary containment.

APPLICABILITY

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen and oxygen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen and oxygen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

If oxygen concentration is ≥ 4.0 v/o at any time while operating in MODE 1, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration

(continued)

BASES

ACTIONS

A.1 (continued)

must be restored to < 4.0 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 4.0 v/o because of the availability of other hydrogen and oxygen mitigating systems (e.g., post-accident nitrogen purge) and the low probability and long duration of an event that would generate significant amounts of hydrogen and oxygen occurring during this period.

B.1

If oxygen concentration cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, power must be reduced to $\leq 15\%$ RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.1.1

The primary containment must be determined to be inerted by verifying that oxygen concentration is < 4.0 v/o. The 7 day Frequency is based on the slow rate at which oxygen concentration can change and on other indications of abnormal conditions (which could lead to more frequent checking by operators in accordance with plant procedures). Also, this Frequency has been shown to be acceptable through operating experience.

REFERENCES

1. Generic Letter 84-09, May 1984.
 2. UFSAR, Section 6.2.5.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses both primary containments and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained, the hatches and blowout panels must be closed and sealed, the sealing mechanisms (e.g., welds, bellows, or O-rings) associated with each secondary containment penetration must be OPERABLE (such that secondary containment leak tightness can be maintained), and all inner or all outer doors in each secondary containment access opening must be closed.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

If secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not

(continued)

BASES

ACTIONS

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

specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring.

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.4

Verifying that one secondary containment access door in each access opening is closed and each equipment hatch is closed and sealed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. In this application, the term "sealed" has no connotation of leak tightness. In addition, for equipment hatches that are floor plugs, the "sealed" requirement is effectively met by gravity. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. In some cases a secondary containment barrier contains multiple inner or multiple outer doors. For these cases, the access openings share the inner door or the outer door, i.e., the access openings have a common inner door or outer door. The intent is to not breach the secondary containment at any

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.4 (continued)

time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times, i.e., all inner doors closed or all outer doors closed. Thus each access opening has one door closed. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency for SR 3.6.4.1.2 has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door status that are available to the operator. The 24 month Frequency for SR 3.6.4.1.4 is considered adequate in view of the existing administrative controls on equipment hatches.

SR 3.6.4.1.3

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. Each SGT subsystem is designed to maintain the secondary containment at ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate of ≤ 4000 cfm. To ensure that all fission products released to the secondary containment are treated, SR 3.6.4.1.3 verifies that a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary can be maintained. When the SGT System is operating as designed, the maintenance of secondary containment pressure cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.3 demonstrates that the pressure in the secondary containment can be maintained ≥ 0.25 inches of vacuum water gauge for 1 hour using one SGT subsystem at a flow rate ≤ 4000 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. The primary purpose of the SR is to ensure secondary containment boundary integrity. The secondary purpose of the SR is to ensure that the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements that serves the primary purpose of ensuring OPERABILITY of the SGT System. This SR need not be performed with each SGT subsystem. The SGT subsystem used for this Surveillance is staggered to ensure that in

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1.3 (continued)

addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. The inoperability of the SGT System does not necessarily constitute a failure of this Surveillance relative to secondary containment OPERABILITY. Operating experience has shown the secondary containment boundary usually passes the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.
 2. UFSAR, Section 15.7.3.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices.

Automatic SCIVs (i.e., dampers) close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations required to be closed during accident conditions are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident (Ref. 2). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIVs is required to ensure that

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

SCIIVs form a part of the secondary containment boundary. The SCIIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated, automatic, isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in the Technical Requirements Manual (Ref. 3).

The normally closed manual SCIIVs are considered OPERABLE when the valves are closed and blind flanges are in place, or open under administrative controls. These passive isolation valves or devices are listed in Reference 3.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIIVs is required.

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 SCIIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

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BASES (continued)

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

The second Note provides clarification, that for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration, and the probability of a DBA, which requires the SCIVs to close, occurring during this short time is very low.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

basis. This is necessary to ensure that secondary containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time of once per 31 days is appropriate because the isolation devices are operated under administrative controls and the probability of their misalignment is low. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows them to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With two SCIVs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIVs to close, occurring during this short time, is very low.

(continued)

BASES

ACTIONS

B.1 (continued)

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

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 to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.4.2.2

Verifying that the isolation time of each power operated, automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is 92 days.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.6.5.
 2. UFSAR, Section 15.7.3.
 3. Technical Requirements Manual.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The SGT System is required by UFSAR, Section 3.1.2.4.12 (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SGT System consists of two fully redundant subsystems that are shared between Unit 2 and Unit 3, each with its own set of ductwork, dampers, charcoal filter train, and controls.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

- a. A demister;
- b. An electric heater;
- c. A rough prefilter;
- d. A high efficiency particulate air (HEPA) filter;
- e. A charcoal adsorber;
- f. A second HEPA afterfilter; and
- g. A centrifugal fan.

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis of the secondary containment. Each SGT subsystem is capable of processing the secondary containment volume, which includes both Unit 2 and Unit 3. The internal pressure of the secondary containment is maintained at a negative pressure of ≥ 0.25 inches water gauge when the SGT System is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building, even at wind speeds of 100 mph.

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BASES

BACKGROUND
(continued)

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, the primary charcoal filter train inlet damper opens, the cooling damper closes, the associated fan starts, and the fan discharge damper opens. When sufficient flow develops, the heater turns on and the flow control damper begins modulating to control system flow and maintain a negative pressure in the secondary containment. If either a low flow or heater off condition exists for the primary subsystem after 20 seconds, the primary subsystem is tripped and the standby SGT subsystem starts.

APPLICABLE
SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Ref. 3 and 4). For these analyzed events, the SGT System is assumed to be manually initiated after 10 minutes to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies 10 CFR 50.36(c)(2)(ii).

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

(continued)

BASES

APPLICABILITY
(continued)

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment...

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT System and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, C.2.2, and C.2.3

During movement of irradiated fuel assemblies, in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action

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BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. Therefore, one SGT subsystem must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of supporting the required radioactivity release control function in MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring the SGT System) occurring during periods where the required radioactivity release control function may not be maintained is minimal.

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BASES

ACTIONS
(continued)

E.1 and E.2

If one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action F.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

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BASES (continued)

SURVEILLANCE
REQUIREMENTSSR 3.6.4.3.1

Operating (from the control room using the manual initiation switch) each SGT subsystem for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

- REFERENCES
1. UFSAR, Section 3.1.2.4.12.
 2. UFSAR, Section 6.5.3.2.
 3. UFSAR, Section 15.6.5.
 4. UFSAR, Section 15.7.3.
 5. Regulatory Guide 1.52, Rev. 2.
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B 3.7 PLANT SYSTEMS

B 3.7.1 Containment Cooling Service Water (CCSW) System

BASES

BACKGROUND

The CCSW System is designed to provide cooling water for the containment cooling heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The CCSW System is operated whenever the containment cooling heat exchangers are required to operate in the suppression pool cooling or containment spray mode of the LPCI System.

The CCSW System consists of two independent and redundant subsystems. Each subsystem is made up of a header, two 3500 gpm pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems is capable of providing the required cooling capacity with two pumps operating to maintain safe shutdown conditions. Also, when available, both subsystems with one pump operating in each subsystem are capable of providing the required cooling capacity to maintain safe shutdown conditions. The two subsystems are separated from each other (and cannot be cross connected), so that failure of one subsystem will not affect the OPERABILITY of the other subsystem. The CCSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The CCSW System is described in the UFSAR, Section 9.2.1, Reference 1.

Cooling water is pumped by the CCSW pumps from the associated crib house suction bay through the tube side of the containment cooling heat exchangers, and discharges to the Service Water (SW) 48-inch discharge header and subsequently to the cooling lake or Illinois River. The normal and ultimate heat sink (UHS) cooling water sources for the CCSW System are described in UFSAR, Section 9.2.5 (Ref. 2). The SW System and the discharge flow paths to the cooling lake and Illinois River are described in UFSAR, Sections 9.2.2 and 2.4.8 (Refs. 3 and 4), respectively.

The system is initiated manually from the control room. If operating and a loss of coolant accident (LOCA) occurs, the

(continued)

BASES

BACKGROUND
(continued)

system is automatically tripped to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time the LOCA signal is manually overridden or clears and adequate electrical power is available.

APPLICABLE
SAFETY ANALYSES

The CCSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the CCSW System to support long term cooling of the primary containment is discussed in UFSAR, Section 6.2.2 (Ref. 5). This analysis explicitly assumes that the CCSW System will provide adequate cooling support to the equipment required for safe shutdown. This analysis includes the evaluation of the long term primary containment response after a design basis LOCA.

Analyses for long term cooling were performed for one LPCI pump, two CCSW pumps, and one containment cooling heat exchanger (this assumes a single failure of an emergency diesel generator) using different pump flow rates and heat exchanger performance values which were also evaluated for the various flows. As discussed in the UFSAR, Section 6.2.1.3.2 (Ref. 6) for these analyses, manual initiation of the OPERABLE CCSW subsystem and containment cooling are assumed to occur 10 minutes after a DBA. The CCSW flow assumed in the analyses is 2500 gpm per pump with two pumps operating in one loop. In this case, the maximum suppression chamber water temperature and pressure are 178.6°F and 9.8 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The CCSW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two CCSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power.

(continued)

BASES

LCO
(continued)

A CCSW subsystem is considered OPERABLE when:

- a. Two pumps are OPERABLE; and
- b. An OPERABLE flow path is capable of taking suction from the UHS and transferring the water to the containment cooling heat exchanger and separately to the associated safety related equipment at the assumed flow rate.

An adequate suction source is not addressed in this LCO since the minimum net positive suction head and maximum suction source temperature are covered by the requirements specified in LCO 3.7.3, "Ultimate Heat Sink (UHS)."

APPLICABILITY

In MODES 1, 2, and 3, the CCSW System is required to be OPERABLE to support the OPERABILITY of primary containment cooling (LCO 3.6.2.3, "Suppression Pool Cooling," and LCO 3.6.2.4, "Suppression Pool Spray"). The Applicability is therefore consistent with the requirements of these systems.

The CCSW System is not required to be OPERABLE in MODES 4 and 5 as it does not support or otherwise affect Shutdown Cooling (SDC) System operation. The Unit 2 CCSW System is required to be OPERABLE during the movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, and during operations with a potential for draining the reactor vessel (OPDRVs). At least one Unit 2 CCSW pump, the Ultimate Heat Sink, and a flow path are required during these conditions to provide backup cooling to the condensing unit of the Control Room Emergency Ventilation Air Conditioning (AC) System (LCO 3.7.4, "Control Room Emergency Ventilation Air Conditioning (AC) System").

ACTIONS

A.1

With one CCSW pump inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE CCSW pumps are adequate to perform the CCSW heat removal function. However, the overall reliability is reduced because a single

(continued)

BASES

ACTIONS

A.1 (continued)

failure in the OPERABLE subsystem could result in reduced CCSW capability. The 30 day Completion Time is based on the remaining CCSW heat removal capability and the low probability of a DBA with concurrent worst case single failure.

B.1

With one CCSW pump inoperable in each subsystem, if no additional failures occur in the CCSW System, then the remaining OPERABLE pumps and flow paths provide adequate heat removal capacity for long term containment cooling to maintain safe shutdown conditions. One inoperable pump is required to be restored to OPERABLE status within 7 days. The 7 day Completion Time for restoring one inoperable CCSW pump to OPERABLE status is based on engineering judgment, considering the level of redundancy provided and the low probability of an event occurring requiring CCSW during this period.

C.1

Required Action C.1 is intended to handle the inoperability of one CCSW subsystem for reasons other than Condition A. The Completion Time of 7 days is allowed to restore the CCSW subsystem to OPERABLE status. With the unit in this condition, the remaining OPERABLE CCSW subsystem is adequate to perform the CCSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE CCSW subsystem could result in loss of CCSW function. The Completion Time is based on the redundant CCSW capabilities afforded by the OPERABLE subsystem and the low probability of an event occurring requiring CCSW during this period.

D.1

With both CCSW subsystems inoperable (e.g., both subsystems with inoperable pumps(s) or flow paths, or one subsystem with an inoperable pump and one subsystem with an inoperable

(continued)

BASES

ACTIONS

D.1 (continued)

flow path), the CCSW System is not capable of performing its intended function. At least one subsystem must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time for restoring one CCSW subsystem to OPERABLE status, is based on the Completion Times provided for the suppression pool cooling and spray functions.

E.1 and E.2

If any Required Action and associated Completion Time of Conditions A, B, C, or D are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual and power operated valve in each CCSW subsystem flow path provides assurance that the proper flow paths will exist for CCSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the CCSW System is a manually initiated system.

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

(continued)

BASES (continued)

- REFERENCES
1. UFSAR, Section 9.2.1.
 2. UFSAR, Section 9.2.5.
 3. UFSAR, Section 9.2.2.
 4. UFSAR, Section 2.4.8.
 5. UFSAR, Section 6.2.2.
 6. UFSAR, Section 6.2.1.3.2.
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