B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The MSSVs and Main Steam Relief Trains (MSRTs) provide overpressure protection for the secondary system. The MSSVs and MSRTs also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the condenser, is not available. This is done in conjunction with the Emergency Feedwater System (EFW) providing cooling water from the EFW Storage Pools.

The MSSVs are spring-loaded safety valves. Each MSSV rated capacity passes 25% of the full steam flow per steam generator at 110% RTP with the valves full open. Two MSSVs are located on each Main Steam Line, outside containment, upstream of the main steam isolation valves and downstream of the MSRT branch line, as described in FSAR Section 10.3 (Ref. 1).

The MSSVs along with the MSRTs provide overpressure protection of the main steam piping and steam generators. Together, the MSSVs and MSRTs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure during an anticipated operational occurrence (AOO) or an accident considered in the design basis accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in FSAR Section 15.2 (Ref. 3). Of these, the closure of a single main steam isolation valve without main steam bypass or partial trip function is the limiting AOO. Closure of a single Main Steam Isolation Valve (MSIV) results in a smaller isolated volume on the secondary side, therefore this event is more limiting than a turbine trip event for secondary system overpressure. The safety analysis demonstrates that the transient response for a single MSIV closure occurring from full power presents no hazard to the integrity of the RCS or the Main Steam System.

BASES

APPLICABLE SAFETY ANALYSES (continued)

This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSRTs and MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. These events are bounded by the single MSIV closure event.

The safety analyses discussed above assume that the low setpoint MSSV of the affected steam generator is out of service and that the MSRT is the single failure.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

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

LCO

The accident analysis requires that the two MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 100.5% RTP. The LCO requires that the two MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the postulated accident analysis.

The OPERABILITY of the MSSVs are defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and be closed or reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

The LCO is modified by a Note. In MODE 4 when a steam generator is relied upon for heat removal, only one MSSV is required to be OPERABLE. Because of the reduced heat removal requirements and the short period of time in MODE 4, one MSSV is sufficient to relieve steam generator overpressure.

BASES

APPLICABILITY

In MODES 1, 2, and 3, two MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODE 4 when a steam generator is relied upon for heat removal, one MSSV per steam generator is required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, and thus cannot be overpressurized; there is no requirement for the MSRTs or MSSVs to be OPERABLE in these MODES.

ACTIONS

A.1 and A.2

With one required MSSV inoperable, the associated MSRT is verified OPERABLE. Verification of MSRT OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSRT is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSRT. If the OPERABILITY of the associated MSRT cannot be verified, however, Condition B must be immediately entered.

An alternative to restoring the inoperable MSSV to OPERABLE status is to reduce power so that the available MSSV and MSRT relieving capacity meets ASME Code, Section III requirements for the power level. Operations may continue, provided the RATED THERMAL POWER is reduced by the application of the following formula:

RTP = Y / Z x 100%

where:

RTP = Reduced power requirement (not to exceed RTP);

Y = Total OPERABLE MSSV and MSRT design relieving capacity per steam generator of 4,266,219 lb/hr with one MSSV inoperable;

Z = Required relieving capacity per steam generator of 5,688,292 lb/hr.

BASES

ACTIONS (continued)

The Completion Time of 4 hours for Required Action A.2 is a reasonable time period to reduce reactor power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs.

B.1 and B.2

If the Required Action and associated Completion Time cannot be met or if two or more MSSVs are inoperable, 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 6 hours, and in MODE 4 without reliance upon a steam generator for heat removal within 24 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code (Ref. 4) requires that safety valve tests shall be performed as required by Appendix I of the ASME OM Code.

The ASME OM Code requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME OM Code specifies the activities and frequencies necessary to satisfy the requirements. The SR allows a ± [3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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

BASES

REFERENCES

1. FSAR Section 10.3.

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

3. FSAR Section 15.2.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located on each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and main steam relief train (MSRT), to prevent MSSV and MSRT isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, turbine bypass, and other auxiliary steam supplies from the steam generators.

The MSIVs are controlled by two redundant and parallel control lines. Each control line is composed of:

a. Two fast closure pilot valves in series actuating a common fast closure distributor; and

b. An exercise pilot valve actuating an exercise distributor.

The arrangement of pilot valves prevents a failure in any pilot valve to cause either a spurious closing (two pilot valves in series) or a failure to close (two distributors in parallel). The MSIVs fail safe position is closed on loss of control or power supply. The pilot valves are de-energized to close the MSIVs.

The MSIVs are closed under faulted conditions by the Distributed Control System. The MSIVs can also be closed manually. The MSIVs fail closed on loss of control or actuation power.

A description of the MSIVs is found in FSAR Section 10.3 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the main steam line break (MSLB) inside containment, discussed in FSAR Section 6.2 (Ref. 2). It is also affected by the accident analysis of the MSLB and feedwater line break events presented in FSAR Chapter 15 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment analysis is the MSLB inside containment, with offsite power available, and failure of the MSIV on the affected steam generator to close. At lower power levels, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIVs contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different MSLB events against different acceptance criteria. The double-ended guillotine break of a main steam line in the valve compartment in the Safeguards Building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB outside containment, upstream of an MSIV at hot zero power is the limiting case for a post trip return to power. The analysis includes a spectrum of break sizes, scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems is delayed. The worse case single failure is a main steam relief train control valve associated with one of the unaffected steam generators failed in the fully open position (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

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

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, two control lines per MSIV are OPERABLE, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.34 (Ref. 4) limits or the NRC staff approved licensing basis.

BASES

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated, when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally most of the MSIVs are closed and the steam generator energy is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With only one control line of one or more MSIVs inoperable in MODE 1, the affected MSIV (s) can still be closed by the other control line, however actions must be taken to restore the inoperable control line(s) to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable considering the MSIV would be closed by the OPERABLE control line in the event of an accident.

B.1

With one MSIV inoperable due to the inoperability of both control lines or reasons other than Condition A, the MSIV must be restored to OPERABLE status within 8 hours. Otherwise 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 MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable based on operating experience to reach MODE 2 and to close the MSIV(s) in an orderly manner without challenging plant systems.

C.1

If Required Action A.1 or B.1 cannot be met within the required Completion Time, 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 MODE 2 within 6 hours and Condition D would be entered. The Completion Times are reasonable based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

D.1 and D.2

Condition D is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is reasonable based on operating experience.

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

E.1 and E.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that each MSIV and its pilot valves are OPERABLE, i.e., that it can be closed on demand. The test is performed one valve at a time using one control line only, in MODES 1 and 2, under stable plant conditions. The Surveillance Frequency of 31 days is consistent with operating experience of similar MSIVs on existing plants.

SR 3.7.2.2

This SR verifies freedom of movement of the valve stem and disk by partial valve closure and re-opening. The MSIV design allows for this test during power operation without impairing power generation and without risk of full valve closure. The Surveillance Frequency of 92 days is consistent with operating experience of similar MSIVs on existing plants.

The Frequency is in accordance with the Inservice Testing Program and is in accordance with the ASME OM Code (Ref. 5). This SR is modified by a Note that limits this surveillance to MODES 1 and 2.

SR 3.7.2.3

This SR verifies that MSIV closure time is within the limit assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. The MSIVs can not be full stroke tested when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME OM Code (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.4

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSIV testing is every 24 months on a STAGGERED TEST BASIS for each control line. As a result, each MSIV will have a closure test on a 24 month Frequency. The test will alternate use of the two redundant control lines. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint. This SR is modified by a NOTE that requires the performance of this surveillance only in MODES 1 and 2.

REFERENCES

1. FSAR Section 10.3.
2. FSAR Section 6.2.
3. FSAR Chapter 15.
4. 10 CFR 50.34.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater (MFW) Valves

BASES

BACKGROUND

On each of the four steam generators (SGs), the Main Feedwater valves (MFW Full Load Isolation Valves (MFWFLIVs), MFW Full Load Control Valves (MFWFLCVs), MFW Low Load Isolation Valves (MFWLLIVs), MFW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Isolation Valves (MFWIVs)) are located in valve compartments, physically separated from each other and from other systems. Within these valve compartments, the MFW lines are arranged in three flow paths, one very low load, one low load, and one full load flow path. The full load flow path for each steam generator includes one MFWFLCV, one MFWFLIV, and the MFWMIV. The low load flow path for each steam generator includes one MWFLLCV, one MFWLLIV, and the MFWIV. The very low load flow path for each steam generator includes one MFWVLLCV, one MFWLLIV, and the MFWIV. Each of these flow paths can be isolated redundantly by one isolation valve, one control valve, or the MFWIV. The MFWLLIV allows isolation of the low load and the very low load flow path at the same time.

The closure of these valves allows limiting the filling of the steam generators in case of a too high feedwater flowrate which could impair the functioning of the safety valves of the Main Steam System.

In the event of a secondary side pipe rupture inside containment, the valves also limit the quantity of high energy fluid that enters containment through the break and provide a pressure boundary for the controlled addition of Emergency Feedwater (EFW) to the intact loops. They also reduce the cooldown effects in case of Main Steam Line Breaks (MSLBs) or in case of excessive increase in feedwater flowrate caused by a Feedwater System malfunction.

The MFWIV outside containment and an MFW check valve inside containment provide isolation of the containment.

The MFWFLIVs and MFWFLCVs close on a reactor trip. The low and very low control and isolation valves close in response to steam generator level as described in Reference 1. The MFW valves may also be actuated manually.

A description of the MFW valves is found in FSAR Section 10.4.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The full load line must be isolated on each of the four SGs by redundant means in case of reactor trip or a high SG level signal. The low and very low load line of the affected SG must be isolated in case of high level, low pressure or a high pressure drop signal coming from the SG. These actions are needed to mitigate the following accidents: MSLB; Feedwater Line Break (FWLB); Steam Generator Tube Rupture (SGTR); or Feedwater Malfunction. The failure of these respective valves to close could lead to an overcooling event causing re-criticality (in case of MSLB or Feedwater Malfunction), to an increase the mass and energy releases inside containment (in case of MSLB or FWLB) or to fill the steam lines with feedwater (in case of SGTR or Feedwater Malfunction).

The MFW valves close on reactor trip and feedwater isolation signals as described in Reference 1. Each flow path has three isolation or control valves in series in addition to a check valve located inside containment.

The MFW valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFW isolation and control valves will reduce or isolate MFW flow to the steam generators, as required, following an excessive feedwater flow accident, a FWLB, an MSLB, or an SGTR. The MFWIV provides isolation for events requiring containment isolation and are addressed by LCO 3.6.3, “Containment Isolation Valves.”

This LCO requires that four MFWFLIVs, four MFWFLCVs, four MFWLLIVs, four MFWLLCVs and four MFWVLLCVs be OPERABLE. The MFWFLIVs, MFWFLCVs, MFWLLIVs, MFWLLCVs and, MFWVLLCVs are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment, the introduction of water into the main steam lines, or an overcooling of the primary circuit depending on the accident considered.

BASES

APPLICABILITY

The MFW isolation and control valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and in the steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1 and in MODES 2 and 3 except when closed and de-activated, the full-load, low load, and very low load isolation and control valves are required to be OPERABLE to limit the amount of water in the steam generator, to limit the overcooling of the primary circuit, or to limit the amount of water that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low and all MFW valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each MFW flow path.

<u>A.1 and A.2</u>

With one valve in one or more full load flow paths inoperable, close or isolate the inoperable affected flow path in 72 hours. When the flow path is isolated, it is performing its required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves; and the low probability of an event that would isolation of the main feedwater flow paths occurring during this period.

For inoperable full load flow path valves that cannot be restored to OPERABLE status within the specified Completion Time but are closed or isolated, the flow paths must be verified on a periodic basis to be closed or isolated. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

BASES

ACTIONS (continued)

B.1

With two inoperable valves in the same full load flow path, there may be no redundant system to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same full load flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected full load valves in each full load flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each full load flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFWFLIV or MFWFLCV, or otherwise isolate the affected flow path.

C.1 and C.2

With one or more MFWLLIVs, MFWLLCVs, or MFWVLLCVs in the low load or very low load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 8 hours or isolate the affected flow path. When the valves are closed, they are performing their required safety function

Inoperable MFW low load and very low load flow path valves that are closed as a result of this Required Action, must be verified on a periodic basis that they are closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

D.1 and D.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV and MFWVLLCV is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. This SR is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME OM Code (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

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

SR 3.7.3.2

This SR verifies that each valve can close on an actual or simulated actuation signal. This Surveillance is normally performed during shutdown or upon returning the plant to operation following a refueling outage.

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

REFERENCES

1. FSAR Section 10.4.7.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.4 Main Steam Relief Trains (MSRTs)

BASES

BACKGROUND

The MSRTs provide overpressure protection for the secondary system. They also provide protection against overpressurization of the reactor coolant pressure boundary (RCPB). The MSRTs also provide a method for cooling the unit to Residual Heat Removal (RHR) System entry conditions should the preferred heat sink via the condenser not be available. This is done in conjunction with the Feedwater or Emergency Feedwater System.

One MSRT is provided for each steam generator, outside containment, on a branch line that is upstream of the Main Steam Safety Valves (MSSVs) and the Main Steam Isolation Valves (MSIVs). The MSRT rated capacity passes 50% of the full steam flow per steam generator at 110% RTP with the valves full open. Each MSRT consists of one Main Steam Relief Control Valve (MSRCV) located downstream of one Main Steam Relief Isolation Valve (MSRIV).

The MSRCVs are motorized control valves, normally open, which allow control of the steam generator steam pressure. The MSRCVs provide a means of controlling MSRT steam flow to prevent overcooling the RCS. The MSRCVs allow mitigation of the effects of a stuck open MSRIV. The MSRCVs are automatically positioned based on THERMAL POWER.

The MSRIVs are angle globe valves with a motive steam-operated piston actuator, operated by two parallel sets of two pilot valves in series. The arrangement of pilot valves prevents a failure in any pilot valve from causing either a spurious opening (two pilot valves in series) or a failure to open (two sets of pilot valves in parallel). The MSRIVs are normally closed, with the pilot valves kept closed (de-energized). The MSRIVs are designed to open quickly and automatically on demand from the Distributed Control System.

A description of the MSRCVs and the MSRIVs is found in FSAR Section 10.3 (Ref. 1)

BASES

BACKGROUND (continued)

Each MSRT minimum required capacity is 50% of the full steam generation of the assigned steam generator, at a design pressure of 1435 psig, thus limiting the system pressure to ≤ 110% of the steam generator design pressure, in order to meet the requirements of the ASME Code, Section III (Ref. 2). The minimum required capacity, combined with the MSSV capacity, provides 100% flow relief at steam generator design pressure per steam generator (SG) at ≤ 110% of the SG design pressure.

Each MSRT maximum capacity is limited to 61% of the full load steam generation of its assigned steam generator, at design pressure of 1435 psig, thus limiting the consequences of MSRIV spurious opening with regards to Reactor Coolant System overcooling and reactivity control.

The MSRTs are actuated automatically by the Distributed Control System, but can be controlled manually by the operator.

APPLICABLE SAFETY ANALYSES

The MSRTs, along with the MSSVs (LCO 3.7.1), are credited in the mitigation of anticipated operational occurrences (AOOs) and postulated accidents in Reference 3. The MSRTs control secondary system pressure to less than the 110% design limit without challenging the MSSVs in the following events:

a. Loss of normal feedwater,

b. Loss of non-emergency AC power,

c. Inadvertent Extra Borating System actuation,

d. Uncontrolled control bank withdrawal at power,

e. Small break LOCA (SBLOCA).

For other analyzed events, an MSRT would normally actuate to control secondary system pressure but is assumed to be the limiting single failure:

a. Inadvertent opening of an MSSV,

b. Main Steam System piping failure,

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Turbine trip,

d. Inadvertent closure of an MSIV,

e. CVCS malfunction that increases RCS inventory,

f. SGTR.

For these events, the remaining MSRTs and/or MSSVs act to control secondary system pressure to less than the 110% design limit.

In the event of an SBLOCA, the secondary side pressure increases rapidly following reactor trip and turbine trip. This pressure increase is controlled by the MSRTs. Following receipt of an SI signal, the MSRTs initiate a partial cooldown which is a controlled secondary system depressurization from 1414.7 psia to 900 psia at a rate corresponding to 180°F/hr.

In the event of an SGTR, the MSRT setpoint for the affected SG is reset to a setpoint above the MHSI delivery pressure, limiting the flow through the tube break. The MSRTs on the intact SGs are used to perform a partial cooldown of the unit at 180°F/hr to 870 psia. The MSRTs are then used for a controlled cooldown at 90°F/hr.

For events analyzed in Reference 3, the MSRTs are credited for long-term decay heat removal once stable plant conditions are achieved.

The MSRTs valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analyses require all four MSRTs to be OPERABLE to provide overpressure protection for AOOs and postulated accidents. The MSRTs are also credited for partial cooldown of the unit following an SGTR or SBLOCA. The LCO requires that all four MSRTs be OPERABLE as assumed in the accident analyses.

The OPERABILITY of the MSRTs are defined as the ability to open upon demand within the setpoint tolerances, to relieve SG overpressure, and to close or reseat when pressure has been reduced. The OPERABILITY of the MSRTs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

BASES

LCO (continued)

This LCO provides assurance that the MSRTs will perform their design safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

The LCO is modified by a Note. In MODE 4 when a steam generator is relied upon for heat removal, only two MSRTs are required to be OPERABLE. Because of the reduced heat removal requirements and the short period of time in MODE 4, two MSRTs are sufficient to relieve steam generator overpressure.

APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is relied upon for heat removal, the MSRTs are required to be OPERABLE to prevent Main Steam System overpressurization and to provide a decay heat removal path in conjunction with the Emergency Feedwater System.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, there are no credible transients requiring the MSRTs. In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, decay heat removal is provided by the Residual Heat Removal System.

ACTIONS

A.1

With one control line inoperable for opening in one or more MSRIVs (i.e., one pilot valve is inoperable for opening), the affected MSRIVs are still OPERABLE, however the control line(s) must be restored to OPERABLE status in 72 hours. This Completion Time is based on the following:

a. Redundancy for MSRIV opening is provided by the second control line.

b. In case of an event with loss of the condenser and assuming a single failure on the second control line of one MSRIV to open, the residual heat removal can still be ensured by the other MSRTs.

c. In case of an overpressure event and assuming a single failure of the second control line of one MSRIV to open which leads to failure to open of the associated MSRIV, the redundancy provided by the two associated OPERABLE MSSVs ensure the pressure limitation in the affected SG.

BASES

ACTIONS (continued)

B.1

In case one pilot valve is open in one or both control lines of one or more MSRIVs, the isolation function of the MSRIV is not assured. The control line(s) must be restored to OPERABLE status in 30 days. This Completion Time is based on redundancy for MSRIV closure provided by the second pilot valve in series and by the MSRCV.

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

With one required MSRT inoperable, the associated MSSVs are verified OPERABLE and action must be taken to reduce power to 50% RTP so that the available MSSV relieving capacity meets ASME Code, Section III requirements for the power level. The MSRT must be restored to OPERABLE status within 72 hours. Verification of MSSVs OPERABILITY is performed as an administrative check by examining logs or other information to determine if MSSVs are out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSSVs. If the OPERABILITY of the associated MSSVs cannot be verified, however, Condition D must be immediately entered.

D.1 and D.2

If Required Action A.1, B.1, or C.1 and C.2 cannot be met within the required Completion Times, 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 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within 24 hours.

With two or more required MSRTs inoperable, the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. 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 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within 24 hours.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies each MSRIV OPERABILITY by opening the valve and then by closing the MSRIV. This SR verifies the valve OPERABILITY in accordance with an owner controlled document controlled under 10 CFR 50.59 and the Inservice Testing Program. The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.4.2

This SR verifies each MSRCV OPERABILITY by stroking the valve through a full cycle. The test can be performed during power operation under stable conditions without impairing power operation because the MSRIV stays closed during the test. The test can also be performed in hot shutdown conditions before plant shutdown. The Frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Distributed Control System during power operation, which ensures that the valve is not blocked in a specific position.

SR 3.7.4.3

This SR demonstrates that each MSRIV actuates on an actual or simulated steam pressure setpoint signal. The 24 month Frequency is based on the need to perform the test during either hot or cold shutdown conditions. The Frequency is reasonable based on the fact that opening a MSRIV is not possible during power operation and on operating experience of similar MSRIVs on existing plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.4 and 3.7.4.5

This SR demonstrates that each MSRCV is automatically positioned based on THERMAL POWER and is switched into SG pressure control mode on an actual or simulated MSRIV opening. The test can be performed in hot shutdown conditions before plant shutdown. The Frequency of once per cycle is reasonable based on operating experience and on the fact that the MSRCV operates under the control of the Distributed Control System during power operation, which ensures that the valve is not blocked in a specific position.

REFERENCES

1. FSAR Section 10.3.
2. ASME Boiler and Pressure Vessel Code, Section III.
3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System. The EFW pumps take suction from their respective EFW storage pool (SP) (LCO 3.7.6) and normally pump to their respective steam generator secondary side via separate and independent connections. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1) or Main Steam Relief Trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

The EFW System consists of four motor driven EFW pumps and four EFW SPs configured into four separate trains. The inventory of the four EFW SPs is available to all EFW pumps through the supply header. The supply header isolation valves and the discharge header isolation valves are maintained closed during normal plant operation and can be opened, as necessary, to change component alignments. The supply header isolation valves can be operated locally. The discharge header isolation valves are motor operated and also have manual hand wheels so that they can be operated from the MCR or locally.

The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each EFW pump is powered from an independent Class 1E power source.

The non-safety Startup and Shutdown System (SSS) is used for supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The EFW System is designed to supply sufficient water to the steam generator(s) (SG) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the EFW System supplies sufficient water to cool the unit to Low Head Safety Injection (LHSI) entry conditions, with steam released through the MSRTs.

The EFW System actuates automatically on low steam generator water level signal generated by the Distributed Control System (LCO 3.3.1). The EFW System also actuates on loss of offsite power signal with safety injection.

The EFW System is discussed in FSAR Section 10.4.9 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The EFW System mitigates the consequences of any event with loss of normal feedwater supplies.

The design basis of the EFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest MSSV set pressure plus 3%.

In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

There are four EFW trains. Each EFW train has a separate SP. All four EFW SPs, the supply and discharge headers, and the four injection paths are required to be OPERABLE. One EFW pump train is assumed to be unavailable due to maintenance and a second EFW pump train or its normal injection pathway is assumed to be lost to a single failure. Note, an EFW pump train includes the pump, discharge check valve, flow control valve, and piping to the manual isolation valves on the suction and discharge of the pump.

The two remaining EFW trains provide sufficient flow for decay heat removal as required by the accident analysis. For certain sized feedwater line breaks, one of the remaining EFW pumps feeds a faulted steam generator. This pump is re-aligned from the MCR at 30 minutes to feed through the injection pathway associated with the EFW train whose pump is unavailable due to maintenance. The reactor coolant pumps in the two unfed loops are tripped.

The limiting accident for the EFW System is a Main Feedwater Line Break (MFWLB).

In addition, the minimum available EFW flow and system characteristics are considerations in the analysis of a small break loss of coolant accident (SBLOCA).

The Distributed Control System automatically actuates the EFW pumps and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power with safety injection.

The EFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii) for operation in MODES 1, 2, and 3, and Criterion 4 of 10 CFR 50.36(c)(2)(ii) for operation in MODE 4.

BASES

LCO

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

Four EFW pumps, the supply and discharge headers, and the four injection paths are required to be OPERABLE to ensure decay heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering each of the pumps from independent emergency buses.

The EFW System is configured into four trains, which share supply and discharge headers. The EFW System is considered OPERABLE when the components and flow paths required to provide redundant EFW flow to the steam generators are OPERABLE. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

In MODE 4 when a steam generator is relied upon for heat removal with two EFW pumps OPERABLE, operation is allowed to continue because only two EFW pumps are required in accordance with the Note that modifies the LCO. Because of the reduced heat removal requirements and the short period of time in MODE 4 when a steam generator is relied upon for heat removal, one EFW pump is sufficient to remove decay heat. The second required pump provides single failure protection.

APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is relied upon for heat removal, the EFW System is required to be OPERABLE in the event that it is called upon to function when MFW or offsite power are lost. In addition, the EFW System is required to supply enough makeup water to replace the secondary inventory, lost as the unit cools to MODE 4 conditions where a steam generator is not relied upon for heat removal.

In MODES 4 when a steam generator is not relied upon for heat removal, 5 and 6, the EFW System is not required.

BASES

ACTIONS

A Note prohibits the application of LCO 3.0.4.b for two or more EFW pump trains inoperable when entering MODE 1. There is an increased risk associated with entering MODE 1 with two or more EFW pump trains inoperable and the provisions of LCO 3.0.4.b, which allows entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1

With one EFW pump train inoperable in MODE 1, 2, or 3, action must be taken to restore OPERABLE status within 120 days. The 120 day Completion Time is reasonable, based on the FSAR Chapter 15 analysis assumption that one EFW pump train is not available due to maintenance.

B.1

With two EFW pump trains inoperable in MODES 1, 2, or 3, action must be taken to restore at least one inoperable EFW pump train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period.

C.1

With a downstream injection pathway or supply or discharge header inoperable in MODE 1, 2, or 3, action must be taken to restore the inoperable pathway or header to OPERABLE status in 8 hours. The 8 hour Completion Time is reasonable, based on redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of a postulated accident occurring during this time period.

BASES

ACTIONS (continued)

D.1 and D.2

When Required Action A, B, or C and associated Completion Time cannot be met; or if three EFW trains are inoperable in MODE 1, 2, or 3; 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 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within 24 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 plant systems.

E.1

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

With four EFW pump trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown and only non-safety means for conducting a cooldown with the SSS. In such a condition, the unit should not be perturbed by any action, including a power change that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one EFW pump train to OPERABLE status.

F.1 and F.2

In MODE 4, either the reactor coolant pumps or the LHSI loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one of the two required EFW pump trains inoperable when a steam generator is relied upon for heat removal, action must be taken to restore an inoperable EFW pump train to OPERABLE status in 72 hours. The 72 hour Completion Time is reasonable based on redundant capabilities of the EFW System.

An alternative to Required Action F.1 is to be in MODE 4 without reliance upon a steam generator for heat removal within an additional 24 hours. The 96 hour Completion Time is reasonable based on redundant capabilities of the EFW System.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

SR 3.7.5.2

Each EFW discharge header cross-connect valve is required to be cycled in order to assure the capability for any EFW pump to feed any steam generator as assumed in the main feedwater line break (Ref. 3) The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.7.5.3

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME OM Code (Ref. 2). Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME OM Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

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

SR 3.7.5.5

This SR verifies that the EFW is properly aligned by verifying the flow path from each storage pool to its respective steam generator prior to entering MODE 2 after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of EFW flow paths must be verified before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the SP to the steam generators is properly aligned.

REFERENCES

1. FSAR Section 10.4.9.
2. ASME Code for Operation and Maintenance of Nuclear Power Plants.
3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater (EFW) Storage Pools

BASES

BACKGROUND

The EFW storage pools (SP) provide a safety-related source of water to respective steam generator (SG) for removing decay and sensible heat from the Reactor Coolant System (RCS). The EFW SPs provide a passive flow of water, by gravity, to the EFW System (LCO 3.7.5). The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) (LCO 3.7.1) or the Main Steam Relief Trains (MSRTs) (LCO 3.7.4). If the main condenser is available, steam may be released via the turbine bypass valves.

The inventory of the four EFW SPs is available to all EFW pumps through the supply header. The supply header isolation valves and the discharge header isolation valves are maintained closed during normal plant operation and can be opened, as necessary, to change component alignments. The supply header isolation valves can be operated locally. The discharge header isolation valves are motor operated and also have manual hand wheels so that they can be operated from the MCR or locally. Because the EFW SPs are principal components in removing residual heat from the RCS, they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The EFW SPs are designed to Seismic Category I to ensure availability of the feedwater supply.

A description of the EFW SPs is found in FSAR Section 10.4.9 (Ref. 1).

The Demineralized Water Distribution System (DWDS), with more than 260,000 gallons available, provides the normal make-up supply to the EFW SPs. The DWDS can be aligned to the EFW SPs from the MCR.

APPLICABLE SAFETY ANALYSES

The EFW SPs provide cooling water to remove decay heat and to cool down the unit following the loss of normal feedwater supplies due to anticipated operational occurrences and accidents addressed in FSAR Chapter 15 (Ref. 3).

The limiting case for sizing of the EFW SPs is a natural circulation cooldown following a LOOP in accordance with BTP 5-4 requirements. A failed closed EFW level control valve is the assumed single failure that results in an unfed SG and stagnant RCS loop.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EFW SPs satisfy the requirements of Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The EFW SPs required usable volume of 365,000 gallons is based on a cooldown to RHR entry conditions for the bounding BTP 5-4 natural circulation cooldown described in FSAR Section 5.4.7.3.2 (Ref. 2). This basis is established in Reference 1 and exceeds the volume required by the accident analysis.

The OPERABILITY of the EFW SPs is determined by summing the available tank volumes. The volume in an SP is considered usable when it is aligned to its respective EFW pump.

APPLICABILITY

In MODES 1, 2, 3, and 4 when a steam generator is relied upon for heat removal, the EFW SPs are required to be OPERABLE to support EFW System OPERABILITY.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, the EFW SPs are not required because the EFW System is not required.

ACTIONS

<u>A.1, A.2, and A.3</u>

With one or more of the EFW SPs inoperable in MODE 1, 2, or 3, or MODE 4 when a steam generator is relied upon for heat removal, action must be taken to declare the associated EFW pump train(s) inoperable and verify the availability of the DWDS by administrative means within 4 hours and once every 12 hours thereafter. The 4 hour Completion Time is reasonable, based on operating experience, to verify the required volume of water. Additionally, verifying the volume of water every 12 hours is adequate to ensure an adequate supply continues to be available. The EFW SPs must be restored to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the volume of availability of the DWDS, and the low probability of an event occurring during this time period requiring the inoperable EFW SP.

BASES

ACTIONS (continued)

B.1 and B.2

If the required action and associated Completion Times 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 6 hours and in MODE 4 without reliance upon a steam generator for heat removal within 24 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the EFW SPs contain the required volume of cooling water. The 24 hour Frequency is based on operating experience and are not used by other systems and that the EFW SPs have no other function than to supply water to the EFW trains. Also, the 24 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the EFW SP levels.

SR 3.7.6.2

This SR verifies every 31 days that the EFW SP supply cross connect valves are closed. This verification ensures that the usable volume in the EFW SP is available to its EFW train to ensure EFW train separation. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned EFW supply cross connect valve is unlikely.

REFERENCES

1. FSAR Section 10.4.9.
2. FSAR Section 5.4.7.3.2.
3. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Service Water (ESW) System and thus to the environment.

The CCW System consists of four separate safety classified trains (1, 2, 3 and 4) corresponding to the four layout divisions (1, 2, 3 and 4) and two separate common headers. One of the common headers (common 1 header) is connected normally either to train 1 or to train 2. The other common header (common 2 header) is connected either to train 3 or to train 4. A set of isolation valves per train can separate each train from the common header and either common header is capable of providing safety related cooling of the reactor coolant pump (RCP) thermal barrier cooling common loop. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal from the Distributed Control System, and all nonessential components are isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in FSAR Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the safety related systems and operational cooling loads to the heat sink via the ESW System. This may be during a normal or post accident cooldown and shutdown.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for two CCW trains to remove the post loss of coolant accident (LOCA) heat load from the In-containment Water Storage Tank (IRWST) by cooling the Low Head Safety Injection System heat exchanger at a maximum CCW temperature of 113°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the minimum performance of the CCW System, respectively. During a unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum temperature of 113°F is assumed. This maintains the IRWST fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the ECCS pumps.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from residual heat removal (RHR) entry conditions (T_{avg} < 350°F), to MODE 5 (T_{avg} ≤ 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR loops operating. Two CCW trains are sufficient to remove decay heat during subsequent operations with T_{avg} ≤ 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

To meet single failure criteria for the RCP thermal barrier cooling function, the load is required to be cooled by a common header which is capable of being connected to two OPERABLE CCW trains. A single failure of a train initiates an automatic system response to transfer the common header to the remaining train.

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

BASES

LCO

The CCW System consists of four trains. Four CCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

A CCW train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

With the exception of the RCP thermal barrier cooling common loop, the isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the Low Head Safety Injection heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A Note has been added to indicate that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

Required Action A.1 is modified by a Note indicating that the Required Action of A.1 is not applicable if CCW trains are inoperable in both common headers. In this condition, the RCP thermal barrier cooling common loop cannot be aligned to common header capable of being connected to two OPERABLE trains.

BASES

ACTIONS (continued)

If one CCW train is inoperable, action must be taken to align the RCP thermal barrier cooling common loop to a common header capable of being supplied by two OPERABLE CCW trains within 72 hrs. In this condition, the CCW System can perform the RCP thermal barrier cooling function given a single failure. The 72 hour Completion Time is reasonable, based on the low probability of a postulated accident occurring during this period.

A.2

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function.

B.1

If two CCW trains are inoperable, action must be taken to restore one train to OPERABLE status within 72 hours. In this condition, the remaining OPERABLE CCW trains are adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE trains, and the low probability of a postulated accident occurring during this period.

C.1 and C.2

If a CCW train cannot be restored to OPERABLE status within the associated Completion Time, 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 6 hours and in MODE 5 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 plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

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

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

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

SR 3.7.7.2

Verifying CCW train leakage is within limits assures an adequate water volume is maintained for each CCW train for cooling SIS loads for 7 days in post-seismic operation with no makeup water source available. The 31 day Frequency is based on the need to perform this Surveillance under normal operating and shutdown conditions for each CCW train. Operating experience has shown that these components usually pass the Surveillance when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

The leakage value of 4 gallons per hour considers the worst case pressure differential resulting from one CCWS train operating with the associated train for the same common header depressurized. This alignment results in the greatest potential seat leakage across the isolated common header switchover valves. If the train leakage surveillance is not within allowable limits for a CCW train, that train and the associated train for the common header will be declared inoperable if the associated train is not already out of service. When two CCW trains are inoperable, one train must be restored within 72 hours in accordance with LCO 3.7.7, Action B.1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The duration of SR 3.7.7.2 testing should be long enough for the installed instrumentation to accurately measure the system losses with considerations to environmental changes in temperatures affecting thermal contraction and expansion of water within the CCWS surge tanks.

SR 3.7.7.3

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

SR 3.7.7.4

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

REFERENCES

1. FSAR Section 9.2.2.

2. FSAR Section 6.2.

B 3.7 PLANT SYSTEMS

B 3.7.8 Essential Service Water (ESW) System

BASES

BACKGROUND

The ESW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the ESW System also provides this function for the associated safety related and nonsafety related systems. The safety related function is covered by this LCO.

The ESW System consists of four separate safety related, cooling water trains. Each train consists of a pump, piping, valving, instrumentation, and mechanical filtration. The system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

Additional information about the design and operation of the ESW System, along with a list of the components served, is presented in FSAR Section 9.2.1 (Ref. 1). The principal safety related functions of the ESW System is the removal of decay heat from the reactor and reactor coolant pump thermal barrier cooling via the Component Cooling Water (CCW) System and removal of operational heat from the emergency diesel generator (EDG).

BASES

APPLICABLE SAFETY ANALYSES	The design basis of the ESW System is for two ESW trains, in conjunction with the CCW System, to remove core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Emergency Core Cooling System pumps. The ESW System also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident. The ESW System, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in FSAR Section 5.4.7 (Ref. 3), entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and RHR loops that are operating. Two ESW trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum ESW System temperature of 95°F occurring simultaneously with maximum heat loads on the system. The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The ESW System consists of four trains. Four ESW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads. An ESW train is considered OPERABLE when the pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.
APPLICABILITY	In MODES 1, 2, 3, and 4, the ESW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ESW System and required to be OPERABLE in these MODES. In MODES 5 and 6, the OPERABILITY requirements of the ESW System are determined by the systems it supports.

BASES

ACTIONS

The actions have two Notes added. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, “AC Sources Operating,” should be entered if an inoperable ESW train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, “RCS Loops MODE 4,” should be entered if an inoperable ESW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

A.1

If one ESW train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE ESW trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore an ESW train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two ESW trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE ESW train are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW trains could result in loss of ESW System function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

C.1 and C.2

If an ESW train cannot be restored to OPERABLE status within the associated Completion Time, 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 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

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

Verifying the correct alignment for manual, power operated, and automatic valves in the ESW flow path provides assurance that the proper flow paths exist for ESW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not 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.

SR 3.7.8.2

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.3

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

REFERENCES

1. FSAR Section 9.2.1.
2. FSAR Section 6.2.
3. FSAR Section 5.4.7.

B 3.7 PLANT SYSTEMS

B 3.7.9 Safety Chilled Water (SCW) System

BASES

BACKGROUND

The SCW System provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the SCW System also provides this function for the associated safety related systems. The safety related function is covered by this LCO.

The SCW System consists of four trains. Each train consists of a chiller refrigeration unit, chilled water pumps (two pumps), surge tank, piping, valving, and instrumentation. Normally open motor operated cross-tie valves interconnect the supply and return of Train 1 with Train 2 and interconnect the supply and return of Train 3 with Train 4. Each SCW System chiller is sized to meet the system load requirements of two divisional trains. Heat is rejected to the system chilled water as it passes through the cooling coils of the system users. This heat is rejected from the system as it is pumped through the train chiller refrigeration units. Trains 1 and 4 reject this energy to ambient via air cooled condensers while Trains 2 and 3 have condensers cooled by the Component Cooling Water (CCW) System. Each refrigeration chiller in the four divisions of the SCWS provides sufficient operating redundancy and flexibility in the event of a compressor failure.

During normal operation, at least one train of the divisional pair is in operation. Either Train 1 or Train 2 chiller provides safety chilled water cooling within Safeguard Building Divisions 1 and 2, and the associated Fuel Building Ventilation System (FBVS) load. Likewise, the chiller from either Train 3 or 4 provides safety chilled water cooling for both Safeguard Divisions 3 and 4 and the associated FBVS load. During normal operation, the cross-tie isolation valves (supply and return for both divisions) are normally open. The non-operating chiller and pump(s) are maintained in standby. This configuration also allows for maintenance on the non-operating chiller and pump(s). If the normal operating train pump or chiller fails, a switchover sequence to the standby train is automatically initiated. A planned switchover of the operating train is manually initiated from the MCR.

BASES

BACKGROUND (continued)

Following a loss of offsite power, previously running SCW trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized. To allow divisional maintenance (e.g., maintenance on emergency diesel generators), the required SCWS safety-related components are alternately fed from the adjacent division to provide adequate cooling of certain safety-related components during a design basis event.

The SCW System operation is discussed in FSAR Section 9.2.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SCW System is to provide chilled water as a heat sink for the Control Room Air Conditioning System and Safeguard Building Ventilation System Electrical Division safety-related HVAC Systems in addition to the LHSI pump motor and seal coolers. This supports maintaining an acceptable environment in the main control room (MCR) and for safety-related equipment in the essential rooms housing Electrical, Instrumentation and Control, Emergency Feedwater System, and CCW System equipment in the Safeguard Buildings as well as supporting the long term operation of the cooled LHSI pumps in the event of an AOO or postulated accident. Cooling of at least two electrical divisions is required in order to ensure the ability of the plant to meet all required safety related functions during any AOO or postulated accident.

A single active failure of a component of the SCW System, with a loss of offsite power, does not impair the ability of the system to perform its design function. The SCW System is designed in accordance with Seismic Category I requirements.

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

LCO

The SCW System consists of four trains. Four SCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove heat loads for all events accompanied by a loss of offsite power and a single failure.

An SCW train is considered OPERABLE when one pump, surge tank, the chiller refrigeration unit with multiple compressors, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SCW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SCW System and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SCW System are determined by the systems it supports.

ACTIONS

A.1

If one SCW train is inoperable, action must be taken to restore OPERABLE status within 30 days. In this condition, the three remaining OPERABLE SCW trains are adequate to perform the heat removal function for continued operation and for postulated accidents.

The 30 day Completion Time is based on the redundant capabilities afforded by the three OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

B.1 and B.2

If the SCW train cannot be restored to OPERABLE status within the associated Completion Time, 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 6 hours and in MODE 5 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 plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

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

Verifying the correct alignment for manual, power operated, and automatic valves in the SCW flow path provides assurance that the proper flow paths exist for SCW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not 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.

Verifying the correct alignment for manual, power operated, and automatic valves in the SCW flow path provides assurance that the proper flow paths exist for SCW System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not 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.

SR 3.7.9.2

The SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the system heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SCW system is slow and is not expected over this time period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies proper automatic operation of the SCW train on an actual or simulated actuation signal. The SCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.9.4

Verifying SCW train leakage is within limits assures an adequate volume of water is maintained for each SCW train for 7 days in post-seismic operation with no make water source available. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

The leakage value of 0.5 gallons per hour considers the worst case pressure difference between one operating SCW train with the SCW Cross-tie Supply and Return valves closed and the opposite SCW train shutdown. The leak test differential pressure across the closed cross-tie valves will be established between a normal operating train with pumps in operation and a shutdown train with pumps secured. This alignment would result in the greatest potential seat leakage across the isolated valves. If the train leakage surveillance is not within allowable limits for a SCW train, that train and the opposite train will be declared inoperable. The duration of SR 3.7.9.4 test should be long enough for the installed instrumentation to accurately measure the system losses with considerations to environmental changes in temperatures effecting the thermal contraction and expansion of water in the SCWS.

REFERENCES

1. FSAR Section 9.2.8.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration (CREF)

BASES

BACKGROUND

The CREF provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, [toxic gases], or smoke.

The CREF consists of two independent, redundant trains that recirculate and filter air in the control room envelope (CRE) and a CRE boundary that limits the in-leakage of unfiltered air. Each CREF train consists of a moisture separator, electric heating coil, prefilter a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), a post-filter, and a fan. Ductwork, dampers, doors, barriers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit for normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the in-leakage of unfiltered air into the CRE will not exceed the in-leakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

BASES

BACKGROUND (continued)

The CREF is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove large particles in the air to prevent excessive loading of the HEPA filters and carbon adsorbers.

----------------------------------REVIEWER'S NOTE----------------------------------
The need for toxic gas isolation state will be determined by the COL applicant.

Actuation of the CREF places the system in [either of two separate states (emergency radiation state or toxic gas isolation state) of] the emergency mode of operation[, depending on the initiation signal]. Actuation of [the system to the emergency radiation state of] the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and carbon filters, and initiates control room pressurization and filtered ventilation of the air supply to the CRE.

Outside air is mixed with recirculated air from the CRE. This air flows through the CREF unit into a common recirculation plenum where it mixes with air pulled from the CRE rooms. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. [The actions taken in the toxic gas isolation state are the same, except that the control room operator switches the CREF to a filtration alignment to minimize any outside air from entering the CRE through the CRE boundary.]

The outside air entering the CRE is continuously monitored by radiation [and toxic gas] detectors. One detector output above the setpoint will cause actuation of the emergency mode [, either the emergency radiation state or toxic gas isolation state, as required]. [The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.]

BASES

BACKGROUND (continued)

One CREF train operating in a filtered alignment at a flow rate of ≤ 4000 cfm (≤ 1000 cfm outside air and 3000 cfm of CRE recirculation air), will pressurize the CRE to ≥ 0.125 inches water gauge relative to all external areas adjacent to the CRE boundary. The CREF operation in maintaining the CRE habitability is discussed in FSAR Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Isolation dampers are arranged in series so the failure of one damper to shut will not result in a breach of isolation. The CREF is designed in accordance with Seismic Category I requirements.

The CREF is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a design basis accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body (5 rem total effective dose equivalent (TEDE)).

APPLICABLE SAFETY ANALYSES

The CREF components are arranged in redundant, safety-related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREF provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in FSAR Chapter 15 (Ref. 2).

The CREF consists of two 100% capacity trains. Each CREF train can be aligned with one of the four air conditioning and recirculation trains. There are only two CREF trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both CREF trains with two of the four of the associated air conditioning and recirculation trains are required to be OPERABLE. One CREF train is assumed to be lost to a single failure. The other train provides 100% of the ventilation to the CRE boundary.

----------------------------------REVIEWER'S NOTE----------------------------------
The need for toxic gas isolation state will be determined by the COL applicant.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CREF provides protection from radiological hazards [, toxic gases,] and smoke to the CRE occupants. Reference 4 discusses protection of the CRE occupants and their ability to control the reactor from the control room or from the remote shutdown panels in the event of a smoke challenge.

The worst case single active failure of a component of the CREF, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

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

BASES

LCO

In the event of a postulated accident, one CREF train is required to provide an adequate supply of filtered air to the CRE. To ensure that this requirement is met, both CREF trains must be OPERABLE. The basis for this approach is that two trains are required to satisfy all design requirements (i.e., one train is needed to mitigate the event and other train is assumed to have a single active failure). The failure of both CREF trains could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body 5 rem TEDE in the event of a large radioactive release.

Each CREF train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREF train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. Prefilters, HEPA filters, and carbon adsorbers, and post-filters are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Heater, ductwork, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREF trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for postulated accidents, and that CRE occupants are protected from [toxic gases and] smoke.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design conditions, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be procedurelized, and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREF trains must be OPERABLE to ensure that the CRE will remain habitable during and following a postulated accident (i.e., LOCA, main steam line break, rod ejection, and fuel handling accident).

In MODE 5 or 6, the CREF is also required to cope with a failure of the Gaseous Waste Processing System.

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

ACTIONS

A.1

With one CREF train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the OPERABLE CREF train is adequate to perform the CRE occupant protection function. However, the overall system reliability is reduced. The 7 day Completion Time is based on the low probability of a postulated accident occurring during this time period, and ability of the remaining train to provide the required capability.

BASES

ACTIONS (continued)

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

----------------------------------REVIEWER'S NOTE----------------------------------
The need for toxic gas isolation state will be determined by the COL applicant.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of postulated accident consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body 5 rem TEDE), or inadequate protection of CRE occupants from [toxic gases or] smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 60 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or [toxic gas] event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a postulated accident, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of postulated accident consequences, and that CRE occupants are protected from radiological hazards [, toxic gas] and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of mitigating actions. The 60 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a postulated accident. In addition, the 60 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

BASES

ACTIONS (continued)

C.1 and C.2

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

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREF train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREF train in the emergency mode. This action ensures that the other train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

----------------------------------REVIEWER'S NOTE----------------------------------
The need for toxic gas isolation state will be determined by the COL applicant.

[Required Action D.1 is modified by a Note indicating to place the system in the toxic gas isolation state.]

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains inoperable, or with the CRE boundary inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

F.1

With both CREF trains inoperable in MODE 1, 2, 3, or 4, the CREF may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations which dry out any moisture accumulated in the carbon adsorber beds from humidity in the ambient air should be performed. Each CREF train must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy. The heater energization time and 31 day Frequency are consistent with Reference 8.

SR 3.7.10.2

This SR verifies that the required CREF filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, carbon adsorber efficiency, minimum flow rate, and the physical properties of the activated carbon. Specific test Frequencies and additional information are discussed in detail in the VFTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREF train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of postulated accident consequences is no more than 5 rem whole body or its equivalent to any part of the body 5 rem TEDE and the CRE occupants are protected from [toxic gases] and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of postulated accident consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Mitigating actions, or compensatory measures, are discussed in Regulatory Guide 1.196, Section 2.7.3, (Ref. 5) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating measures as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis postulated accident consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

BASES

REFERENCES

1. FSAR Section 9.4.

2. FSAR Section 15.6.

3. Deleted.

4. FSAR Section 9.5.

5. Regulatory Guide 1.196, Rev. 1, January 2007.

6. NEI 99-03, "Control Room Habitability Assessment," June 2001.

7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2005, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability" (ADAMS Accession No. ML040300694).

8. Regulatory Guide 1.52, Rev. 3, June 2001.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning System (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control room envelope (CRE) following isolation of the control room.

The CRACS operates in the recycling mode with fresh outside air makeup. There are two 100% capacity identical fresh air intake trains. Train 1 is located in Safeguard Building 2 and train 4 is located in Safeguard Building 3. For each intake train, the fresh air is taken from outside environment through a motor-operated inlet isolation damper, and a pressure control damper. If operating in the unfiltered bypass alignment (intake air bypasses the CREF), outside air flows through a prefilter and duct heater. The fresh outside air then goes to the common recirculation plenum and mixes with CRE recycled air. The mixed air is then directed through two of the four air conditioning trains.

During normal and emergency operation each CRACS cooling unit provides 50% of the cooling for the rooms within the CRE. Each CRACS air handling unit is capable of cooling up to 50% of the normal and emergency cooling load to allow two CRACS air handling units to cool the CRE rooms during a station blackout (SBO) event. During an SBO event, the CRACS air handling units will prevent the CRE room temperature from exceeding 78°F.

Each air conditioning train consists of a final filter, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, and backdraft damper. The conditioned air is supplied to the control room envelope (CRE) areas. Electric heaters are installed in the supply air ducts to maintain individual room temperature. The air is pulled from the CRE areas into a common recirculation air plenum and then recycled through the air conditioning units for each train. Upon receipt of a high radiation alarm or upon receipt of a containment isolation alarm, the CREF unit operates in the filtered alignment. Operation of either CREF unit or closure of either outside inlet air isolation damper will shut down the normal kitchen or restroom exhaust fan and close isolation dampers in ducting routed to the safeguard building ventilation system (SBVS) where it is exhausted to the outside environment.

During normal operation, the CREF units operate in the bypass alignment (air bypasses the iodine filtration unit). CRE room exhaust from clean areas continues to recycle back to the recirculation plenum and CRACS air conditioning units.

BASES

BACKGROUND (continued)

Two out of four CRACS Air Conditioning trains operating in the recirculation mode with fresh outside makeup air will provide the required temperature in the Main Control Room (MCR) between 68°F to 78°F.

The CRACS operation in maintaining the CRE temperature is discussed in FSAR Section 9.4.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain the CRE for 30 days of continuous occupancy.

There are four CRACS trains with two trains normally in operation. During emergency operation, one train is assumed to be out for maintenance and a second train is assumed lost to single failure. The two OPERABLE CRACS trains maintain the MCR temperature between 68°F to 78°F. Redundant detectors and controls are provided for control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the CRE, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

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

LCO

Four independent and redundant trains of the CRACS are required to be OPERABLE to ensure that at least two are available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in all four trains. These components include the heating and cooling coils, moisture separators, and associated temperature control instrumentation. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained.

BASES

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

ACTIONS

A.1

With one CRACS train inoperable, the inoperable train must be returned to OPERABLE status within 120 days. The 120 day Completion Time is based on the assumption that the one CRACS train is out of service for maintenance and is consistent with the dose analysis assumptions in FSAR Chapter 15.

B.1

With two CRACS trains inoperable, action must be taken to restore one CRACS train to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS trains are adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in one of the OPERABLE CRACS trains could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or non-safety related cooling means are available.

C.1 and C.2

If any Required Action and Associated Completion Time of Condition A or B is not met in MODE 1, 2, 3, or 4, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

ACTIONS (continued)

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train(s) cannot be restored to OPERABLE status within the required Completion Time, an OPERABLE CRACS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with three or more CRACS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

F.1

If three or more CRACS trains are inoperable in MODE 1, 2, 3, or 4, the CRACS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the control room heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.
REFERENCES	1. FSAR Section 9.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.12 Safeguard Building Controlled Area Ventilation System (SBVS)

BASES

BACKGROUND

The SBVS provides a protected environment in the hot mechanical areas of Safeguards Building Divisions 1, 2, 3, and 4 and the fuel building. The SBVS also filters airborne radioactive particulates from the areas of the active Emergency Core Cooling System (ECCS) components during a Loss of Coolant Accident (LOCA).

The conditioned air supply to all four Safeguard Building Divisions is provided independently for each division by the Electrical Division of Safeguard Ventilation System (Ref. 1). The SBVS supplies the conditioned air for ventilation through a volume control damper and two isolation dampers for each division to the hot mechanical areas of the four Safeguard Building Divisions. The SBVS air supply and exhaust flows are designed to prevent spread of airborne contamination and to maintain a negative pressure in the safeguard building controlled areas and fuel building.

Under normal plant operation, the operational air exhaust from each hot area is drawn independently through a volume control damper and two isolation dampers located on the operational exhaust duct system for each safeguard building. The main exhaust duct of each division is connected to a common concrete duct which runs inside the annulus. The operational air exhaust is then drawn through a concrete duct cell for processing by the normal filtration train of the Nuclear Auxiliary Building Ventilation System prior to release through the vent stack (Ref. 2).

During conditions in which a release of airborne contamination from any of the four hot mechanical areas occurs, the SBVS will redirect the accident air exhaust independently via four separate exhaust lines which join into one common leak-tight exhaust duct inside the annulus. The exhaust duct then connects to an accident exhaust filtration train located in the fuel building. There are two 100% capacity accident iodine exhaust filtration trains in parallel configuration. Each train consists of inlet motor controlled damper, moisture separator, electric heater, upstream HEPA filter, iodine filter with activated carbon, downstream HEPA filter, outlet motor controlled damper, exhaust fan, and non-return damper. The accident air exhaust is processed through one or both independent iodine filtration trains prior to release through the vent stack. The downstream

BASES

BACKGROUND (continued)

HEPA filter is not credited in the analysis, but serves to collect carbon particles and provides a backup in case the upstream HEPA filter bank fails. The moisture separator removes any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and carbon adsorbers.

In case of a LOCA with assumed ECCS leakage, the accident air exhaust from the safeguard building controlled areas and fuel building is also directed through the accident iodine exhaust filtration trains prior to release through the vent stack.

The SBVS accident iodine filtration train is a standby system which may also be operated during normal plant operations. Upon receipt of an actuating signal, the normal air exhaust from the buildings is isolated and the accident air is redirected through the iodine filtration train.

The SBVS is discussed in FSAR Section 9.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The SBVS design basis is established by the consequences of the limiting postulated accident, which is a LOCA with assumed ECCS leakage. The analysis of a LOCA, given in Reference 4, assumes ECCS leakage to the safeguard building controlled areas and fuel building is a conservative four gallons a minute. The SBVS consists of two 100% capacity iodine filtration trains in parallel configuration. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both sets of iodine filtration trains are required to be OPERABLE. One SBVS train is then assumed to be lost due to a single failure. The postulated accident analysis assumes that two trains of the SBVS are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the one train of this filtration system. The amount of fission products available for release from the safeguard building controlled areas and fuel building is determined for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 5).

The SBVS is not credited in the Fuel Handling Accident evaluation.

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

BASES

LCO

Two independent and redundant trains of SBVS Accident Exhaust Filtration are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power.

The failure of both trains could result in the atmospheric release from the safeguard building controlled areas and fuel building exceeding the 10 CFR 50.34 (Ref. 6) limits in the event of a LOCA.

The SBVS Accident Exhaust Filtration train is considered OPERABLE when it's associated:

a. Fan is OPERABLE;

b. Prefilter, HEPA filter and carbon adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

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

The LCO is modified by a Note allowing the safeguard building controlled areas and fuel building boundaries to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for safeguard building or fuel building isolation is indicated.

APPLICABILITY

In MODE 1, 2, 3, or 4, the SBVS Accident Exhaust Filtration train is required to be OPERABLE to provide fission product removal associated with the leakage inside the controlled areas of the safeguard buildings.

In MODE 5 or 6, the SBVS Accident Exhaust Filtration train is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

BASES

ACTIONS

A.1

With one SBVS Accident Exhaust Filtration train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the SBVS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable SBVS train, and the remaining SBVS train providing the required protection.

B.1

----------------------------------REVIEWER'S NOTE----------------------------------
Adoption of Condition B is dependent on a commitment from the licensee to have guidance available describing compensatory measures to be taken in the event of an intentional and unintentional entry into Condition B.

----------------------------------REVIEWER'S NOTE----------------------------------
The need for toxic gas isolation state will be determined by the COL applicant.

If the safeguard building controlled areas or fuel building boundary is inoperable in MODE 1, 2, 3, or 4, the SBVS trains may not be able to perform their intended functions. Actions must be taken to restore an OPERABLE safeguard building controlled areas and fuel building boundaries within 24 hours. During the period that the safeguard building controlled areas or fuel building boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19 and 10 CFR Part 100 should be utilized to protect plant personnel from potential hazards such as radioactive contamination, [toxic gases,] smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a postulated accident occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the safeguard building controlled areas or fuel building boundary.

BASES

ACTIONS (continued)

C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both SBVS Accident Exhaust Filtration trains are inoperable for reasons other than an inoperable safeguard building controlled areas or fuel building boundary (i.e., Condition B), 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 MODE 3 within 6 hours and in MODE 5 within 36 hours. The 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Verifying that safeguard building controlled areas and fuel building negative pressure is within limit ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to safeguards building and fuel building pressure variations and pressure instrument drift during the applicable MODES.

SR 3.7.12.2

Maintaining safeguard building controlled areas and fuel building OPERABILITY requires verifying each access opening door is closed. However, all safeguard building controlled areas and fuel building 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 of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the carbon from humidity in the ambient air. Systems with heaters must be operated for ≥ 15 minutes with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available. The heater energization time and 31 day Frequency are consistent with Reference 7.

SR 3.7.12.4

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

SR 3.7.12.5

This SR verifies that each SBVS train starts and operates on an actual or simulated actuation signal. The 24 month Frequency is consistent with Reference 7.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.6 and 3.7.12.7

The SBVS exhausts the safeguard building controlled areas and fuel building atmosphere to the environment through appropriate treatment equipment. Each safety SBVS train is designed to draw down the safeguard building controlled areas and fuel building to a pressure of ≤ -0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the safeguard building controlled areas and fuel building at a pressure of ≤ -0.25 inches wg at a flow rate ≤ 2,640 cfm from the safeguard building controlled areas and fuel building. To ensure that all fission products released to the safeguard building controlled areas and fuel building are treated, SR 3.7.12.6 and SR 3.7.12.7 verify that a pressure in the safeguard building controlled areas and fuel building that is less than the lowest postulated pressure external to the safeguard building controlled areas and fuel building boundaries can be established and maintained. When the SBVS is operating as designed, the establishment and maintenance of safeguard building controlled areas and fuel building pressure cannot be accomplished if the safeguard building controlled areas or fuel building boundaries is not intact. Establishment of this pressure is confirmed by SR 3.7.12.6. SR 3.7.12.7 demonstrates that the safeguard building controlled areas and fuel building can be maintained at a pressure of ≤ -0.25 inches wg. The primary purpose of these SRs is to ensure safeguard building controlled areas and fuel building boundary integrity. The secondary purpose of these SRs is to ensure that the SBVS train being tested functions as designed. These SRs need not be performed with each safety SBVS train. The SBVS train used for these Surveillances is staggered to ensure that in addition to the requirements of LCO 3.7.12, either safety SBVS train will perform this test. The inoperability of the SBVS does not necessarily constitute a failure of these Surveillances relative to the safeguard building controlled areas and fuel building OPERABILITY. Operating experience has shown the safeguard building controlled areas and fuel building boundaries usually pass these Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.7.12.8

This SR verifies that the SBVS recirculation coolers that cool the hot mechanical areas are capable of removing the design heat load assumed in the safeguards building heat load calculation. This SR consists of a combination of testing and calculations. The 24-month Frequency is appropriate since significant degradation of the SBVS is slow and is not expected over this time period.

BASES

REFERENCES

1. FSAR Section 9.4.6.
2. FSAR Section 9.4.3.
3. FSAR Section 9.4.5.
4. FSAR Chapter 15.
5. Regulatory Guide 1.25, March 1972.
6. 10 CFR 50.34.
7. Regulatory Guide 1.52, Rev. 3, June 2001.

B 3.7 PLANT SYSTEMS

B 3.7.13 Safeguards Building Ventilation System Electrical Division (SBVSED)

BASES

BACKGROUND

The SBVSED provides temperature control for the electrical and instrumentation and control rooms of each safeguards building.

The SBVSED consists of four independent trains that provide cooling and heating of the electrical equipment areas of each safeguards building. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for temperature control. The SBVSED can be operated with or without recycled air depending on the outside air temperature.

The SBVSED is an emergency system which also operates during normal unit operations and accident conditions to provide ventilation and cooling in the electrical equipment areas of the safeguards buildings.

Following a loss of offsite power, previously running SBVSED trains return to operation once the emergency diesel generator is started and the associated AC electrical power division is re-energized.

The SBVSED operation in maintaining the safeguards building temperature is discussed in FSAR Section 9.4.6 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the SBVSED is to provide ventilation and air conditioning to the electrical equipment area of the safeguards buildings following any Design Basis Accident (DBA). There are four SBVSED trains, with one train normally in operation in each of the four safeguards buildings. During emergency operation, one train is assumed to be lost to single failure of a diesel generator. The three OPERABLE SBVSED trains maintain their respective safeguards building in a pre-determined temperature range. The SBVSED is designed in accordance with Seismic Category I requirements. The SBVSED is capable of removing sensible and latent heat loads from the safeguards building, which include consideration of equipment heat loads, to ensure equipment OPERABILITY.

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

BASES

LCO

Four independent trains of the SBVSED are required to be OPERABLE and in operation to ensure that at least three are available, assuming a single failure disabling one train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The SBVSED is considered to be OPERABLE when the individual components necessary to maintain the safeguards building temperature are OPERABLE and in operation in all four trains. These components include the cooling coils and associated temperature control instrumentation.

APPLICABILITY

In MODES 1, 2, 3, 4, the SBVSED must be OPERABLE and in operation to ensure that the safeguards building electrical equipment areas will not exceed equipment operational requirements following a DBA.

In MODES 5 and 6, the OPERABILITY requirements of the SBVSED is determined by the systems that it supports.

ACTIONS

A.1

With one SBVSED train inoperable or not in operation, action must be taken to restore OPERABLE status within 72 hours. In this condition, the remaining OPERABLE SBVSED trains are adequate to maintain the three remaining safeguards building temperature within limits. A non-safety maintenance train is available to provide temperature control in the affected safeguards building electrical area. However, the overall reliability is reduced because a loss of offsite power would result in loss of SBVSED function in the affected train. The 72 hour Completion Time is based on the low probability of an event occurring, the consideration that the remaining safeguards trains can provide the required safety function, and that alternate, non-safety related cooling means are available.

B.1 and B.2

If any Required Action and associated Completion Time of Condition A is not met, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Each SBVSED train is verified to be in operation at a Frequency of 24 hours to verify that ventilation and air conditioning to the electrical equipment area of each safeguard building. The 24 hour Frequency is appropriate since the train is normally in operation and other indications are available to alert the control room to a failure of a SBVSED train.

SR 3.7.13.2

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the in the safeguards building heat load calculation. This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the SBVSED is slow and is not expected over this time period.

SR 3.7.13.3

This SR verifies proper automatic operation of the SBVSED train on an actual or simulated actuation signal. The SBVSED System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR Section 9.4.6.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel 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. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel storage pool design is given in FSAR Section 9.1.2 (Ref. 1). A description of the Fuel Pool Cooling and Purification System is given in FSAR Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in FSAR Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is well within the limits of Table 6 of Regulatory Guide 1.183 (Ref. 5).

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criteria 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

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

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.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel storage pool 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 appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR Section 9.1.2.
2. FSAR Section 9.1.3.
3. FSAR Section 15.7.4.
4. Regulatory Guide 1.25, March 1972.
5. Regulatory Guide 1.183, Table 6, July 2000.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Boron Concentration and Enrichment

BASES

BACKGROUND

The water in the spent fuel pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel assemblies in the spent fuel storage racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 1.0, including uncertainties and tolerances. 10CFR50.68(b)(4) specifies a limiting k_{eff} of 1.0 for normal storage in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains a subcritical condition for the allowed loading patterns (Ref. 1). ANSI/ANS-57.2-1983 (Ref. 2) requires that the criticality safety analyses demonstrate that criticality could not occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences. Therefore, credit is allowed for soluble boron under abnormal and accident conditions since only a single accident scenario need be considered at a time.

APPLICABLE SAFETY ANALYSES

Abnormal or accident scenarios can be postulated in the spent fuel pool that could have more than a negligible positive reactivity effect. An example is the misplacement of a fuel assembly with the maximum allowed enrichment in specific locations in the Region 2 storage racks. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during this event by a minimum boron concentration of 582 ppm enriched boron established by the criticality analysis (Ref. 3). Compliance with the LCO minimum enriched boron concentration limit of 1700 ppm results in the credited concentration being available.

The concentration and enrichment of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The spent fuel pool enriched boron concentration is required to be ≥ 1700 ppm enriched boron. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 1 and 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

BASES

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration or enrichment of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration or enrichment of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pool fuel assembly locations, to ensure proper locations of the fuel assembly, can be performed. However, prior to resuming movement of fuel assemblies, the concentration and enrichment of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel assembly movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR and SR 3.7.13.2 are met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

SR 3.7.15.2

Verification every 24 months that the B-10 enrichment is ≥ 37%ensures that the B-10 concentration assumed in the accident analyses is available. Since the boron in the spent fuel pool is not exposed to a significant neutron flux, 24 months is considered conservative.

REFERENCES

1. FSAR Section 9.1.1.
2. ANSI/ANS-57.2-1983.
3. TN-Rack.0101, Revision 0, "US EPR New and Spent Fuel Storage Rack Technical Report," AREVA Transnuclear Inc., December 2009.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

BASES

BACKGROUND

The spent fuel storage pool as described in Ref. 1 consists of 17 modular racks of various sizes. The spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 382 storage locations, is designed to accommodate new fuel assemblies with a maximum nominal enrichment of 5.0 wt% U-235 or spent fuel assemblies regardless of the discharge burnup. Region 2, with a maximum of 865 storage locations is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup specified in Figure 3.7.16-1. No burnup or enrichment uncertainty is included in Figure 3.7.16-1. Enrichment uncertainty is fuel batch dependent. Burnup uncertainty is controlled by plant administrative procedures via either a reduction in the calculated assembly burnup or an increase in the burnup limits determined by analysis. Additionally, new or irradiated fuel assemblies with a combination of burnup and initial enrichment that are not in the acceptable range of Figure 3.7.16-1 may be stored in Region 2 in a checkerboard arrangement (rack location is surrounded on all four adjacent faces by empty rack cells or other non-reactive materials). The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.16.

The uncertainty for assembly burnup can be based on the $F_{\Delta H}$ uncertainty, which is derived from the maximum pin power uncertainty at any time during the cycle. The $F_{\Delta H}$ uncertainty can be applied to reduce the assembly burnup throughout the cycle to provide a conservative low estimate of the burnup for the spent fuel pool. This method assumes that the maximum pin power uncertainty difference is maintained in the same assembly during the entire cycle, which is not realistic and results in overly conservative fuel assembly burnups.

If more accuracy is desired, measured to predicted comparisons of burnup (time integrated power) using the Aeroball Measurement System (AMS) can be used to define a more accurate uncertainty. Time integrated calculated power and plant measured power can be compared to obtain a calculated burnup uncertainty. Using multiple maps throughout the cycles, measured assembly burnups are accumulated using the measured power distribution. Using the same burnup steps, calculated burnups are also derived using the calculated power distribution. The observed difference between calculated and measured burnups is considered to be an adequate measure of the burnup uncertainty using the AMS. This error can be statistically combined with the uranium loading uncertainty and the core thermal power to obtain the overall fuel assembly burnup uncertainty.

BASES

BACKGROUND (continued)

The water in the spent fuel pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel assemblies in the spent fuel storage racks, the design basis for preventing criticality outside the reactor is that there is a 95 percent probability at a 95 percent confidence level, without soluble boron, that the k_{eff} remains below 1.0 (subcritical). A discussion of how soluble boron is credited for the storage of fuel assemblies in the spent fuel pool is contained in the Background for TS 3.7.15. Storage configurations are defined in the criticality analyses (Ref. 2) with a resulting k_{eff} less than 1.0 with no soluble boron under normal storage conditions, including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accidental reactivity insertion. A soluble boron concentration of 582 ppm enriched boron is required to maintain k_{eff} less than or equal to 0.95 for allowable storage configurations, which is well within the 1700 ppm enriched boron requirement of LCO 3.7.15.

Prior to movement of an assembly, it is necessary to verify that SR 3.7.15.1 is current.

APPLICABLE SAFETY ANALYSES

Hypothetical accidents can only take place during or as a result of the movement of a fuel assembly (Ref. 1). For these accident occurrences, the presence of soluble boron in the spent fuel pool (controlled by LCO 3.7.15, "Spent Fuel Pool Boron Concentration and Enrichment") prevents criticality in both regions. By closely controlling the movement of each fuel assembly and by checking the location of each fuel assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.16, maintains the k_{eff} of the spent fuel pool < 1.0, assuming the pool to be flooded with unborated water and ≤ 0.95 with a boron concentration ≥ 582 ppm enriched.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with the LCO, the immediate action is to initiate action to move the noncomplying fuel assembly to the correct location.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the planned spent fuel pool location is acceptable for the fuel assembly being stored as specified in the accompanying LCO.

REFERENCES

1. FSAR Section 9.1.2

2. TN-Rack.0101, Revision 0, “US EPR New and Spent Fuel Storage Rack Technical Report,“ AREVA Transnuclear Inc., December 2009.

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit bounds the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.12, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.15, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 50.34 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in FSAR Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and Main Steam Relief Trains (MSRTs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and MSRTs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a postulated accident to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a postulated accident.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, 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 6 hours, and in MODE 5 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 50.34.

2. FSAR Chapter 15.

B 3.7 PLANT SYSTEMS

B 3.7.18 Main Steam Line Leakage

BASES

BACKGROUND

A limit on leakage from the main steam line inside containment is required to limit system operation in the presence of excessive leakage. Leakage is limited to an amount which would not compromise safety consistent with the Leak-Before-Break (LBB) analysis discussed in Reference 1. This leakage limit ensures appropriate action can be taken before the integrity of the lines is impaired.

LBB is an argument which allows elimination of design for dynamic load effects of postulated pipe breaks. The fundamental premise of LBB is that the materials used in nuclear plant piping are strong enough that even a large throughwall crack leaking well in excess of rates detectable by present leak detection systems would remain stable, and would not result in a double-ended guillotine break under maximum loading conditions. The benefit of LBB is the elimination of pipe whip restraints, jet impingement effects, subcompartment pressurization, and internal system blowdown loads.

As described in Reference 1, LBB has been applied to the main steam line pipe runs inside containment. Hence, the potential safety significance of secondary side leaks inside containment requires detection and monitoring of leakage inside containment. This LCO protects the main steam lines inside containment against degradation, and helps assure that serious leaks will not develop. The consequences of violating this LCO include the possibility of further degradation of the main steam lines, which may lead to pipe break.

APPLICABLE SAFETY ANALYSES

The safety significance of plant leakage inside containment varies depending on its source, rate, and duration. Therefore, both detection and monitoring of plant leakage inside containment are necessary. This is accomplished via the instrumentation required by LCO 3.4.14, “RCS Leakage Detection Instrumentation,” and the RCS water inventory balance (SR 3.4.12.1). Subtracting RCS leakage as well as any other identified non-RCS leakage into the containment area from the total plant leakage inside containment provides qualitative information to the operators regarding possible main steam line leakage. This allows the operators to take corrective action should leakage occur which is detrimental to the safety of the facility and/or the public. A local humidity detection system (Reference 2) also provides an indication of main steam line leakage. The main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria.

BASES

LCO

Main steam line leakage is defined as leakage inside containment in any portion of the four (4) main steam line pipe walls. Up to 1.0 gpm of leakage is allowable because it is below the leak rate for LBB analyzed cases of a main steam line crack twice as long as a crack leaking at ten (10) times the detectable leak rate under normal operating load conditions. Violation of this LCO could result in continued degradation of the main steam line.

APPLICABILITY

Because of elevated Main Steam System temperatures and pressures, the potential for main steam line leakage is greatest in MODES 1, 2, 3, and 4 when a steam generator is relied upon for heat removal.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, a main steam line leakage limit is not provided because the Main Steam System pressure is greatly reduced from normal operating pressure, resulting in lower stresses and a reduced potential for leakage. In addition, the steam generators are not the primary method of RCS heat removal in MODES 5 and 6.

ACTIONS

A.1 and A.2

With main steam line leakage in excess of the LCO limit, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be placed in MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the main steam line pressure and leakage, and also reduces the factors which tend to degrade the main steam lines. The Completion Time of 6 hours to reach MODE 3 from full power without challenging plant systems is reasonable based on operating experience. Similarly, the Completion Time of 36 hours to reach MODE 5 without challenging plant systems is also reasonable based on operating experience. In MODE 5, the pressure stresses acting on the main steam line are much lower, and further deterioration of the main steam line is less likely.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.7.18.1</u> Verifying that main steam line leakage is within the LCO limit assures that the integrity of those lines inside containment is maintained. A local humidity detection system (Reference 2) provides an indication of main steam line leakage. Also, an early warning of main steam line leakage is provided by the automatic system which monitors the containment sump level. Main steam line leakage would appear as unidentified leakage inside containment via this system.
REFERENCES	1. FSAR Section 3.6.3. 2. FSAR Section 7.1.1.

B 3.7 PLANT SYSTEMS

B 3.7.19 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for the removal of process and operating heat from safety related components during an anticipated operational occurrence (AOO) or postulated accident. During normal operation, and a normal shutdown, the UHS also provides this function for the associated safety related and non safety related systems. The safety related function is covered by this LCO.

The UHS consists of four separate safety related, cooling water trains. Each train consists of one mechanical draft cooling tower, associated basin, piping, valving, and instrumentation. Each safety related 2-cell Seismic Category I mechanical draft cooling tower rejects energy from the essential service water (ESW) fluid to ambient and returns the cooled fluid to the UHS cooling tower basin, from which the ESW pumps take suction. Each UHS cooling tower basin is sized for 3 days of post loss of coolant accident (LOCA) operation and ensures adequate volume for the required net positive suction head (NPSH) for the associated ESW pump. Post LOCA evaporative losses are replenished by a safety related Seismic Category I source of makeup water. The train associated safety related make-up source delivers water to each basin at ≥ 300 gpm to maintain the NPSH for the ESW pump for up to 30 days following a LOCA.

The mechanical draft cooling towers and basins are safety related, Seismic Category I structures sized to provide heat dissipation for safe shutdown following an accident. The cooling tower is protected from tornado missiles.

[The Seismic Category I makeup necessary to support 30 days of post accident mitigation is site specific and details are to be provided by the COL applicant.]

Additional information about the design and operation of the UHS is presented in FSAR Section 9.2.5 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the UHS is for two UHS trains, in conjunction with the ESW and Component Cooling Water (CCW) Systems, to dissipate core decay heat and support containment cooling following a design basis LOCA as discussed in FSAR Section 6.2 (Ref. 2). This maintains the In-containment Water Storage Tank fluid within acceptable limits following a LOCA as it is supplied to the Reactor Coolant System by the Safety Injection System pumps. The UHS also provides cooling to the train EDG during an anticipated operational occurrence (AOO) or postulated accident.

The UHS, in conjunction with the ESW and CCW Systems, also cools the unit from residual heat removal (RHR), as discussed in FSAR Section 5.4.7 (Ref. 3), entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of ESW, CCW and RHR loops that are operating. Two UHS trains are sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum UHS temperature of 95°F occurring simultaneously with maximum heat loads on the system.

Each UHS basin is sized for 3 days of post LOCA operation without requiring makeup. UHS basin makeup is required to maintain NPSH for the ESW pumps beyond 3 days. This volume of water is assumed to be at ≤ 90°F during normal plant operation to prevent exceeding the maximum ESW temperature during a LOCA.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure. The UHS cooling tower and basin is designed in accordance with Regulatory Guide 1.27 (Ref. 4), which requires a 30 day supply of cooling water in the UHS basin, or equivalent make-up.

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

BASES

LCO

The UHS consists of four trains. Four UHS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads.

A UHS train is considered OPERABLE when two cooling tower fans, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the UHS basin contains ≥ 23.75 feet of water at ≤ 90°F with capability from makeup from OPERABLE source. [COL applicant to provide definition of OPERABLE makeup source.]

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS is determined by the systems it supports.

ACTIONS

A.1

If one UHS train is inoperable, action must be taken to restore OPERABLE status within 120 days. In this condition, the remaining OPERABLE UHS trains are adequate to perform the heat removal function.

The 120 day Completion Time to restore a UHS train to OPERABLE is reasonable since its operation is not assumed in the safety analysis to mitigate the consequences of postulated accidents or AOOs, it provides a reasonable time for repairs, and the low probability of a postulated accident or AOO occurring during this period.

B.1

If two UHS trains are inoperable, action must be taken to restore one to OPERABLE status within 72 hours. In this condition, the two remaining OPERABLE UHS trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE UHS trains could result in loss of UHS function.

The 72 hour Completion Time is based on the redundant capabilities afforded by the two OPERABLE trains, and the low probability of a postulated accident occurring during this time period.

BASES

ACTIONS (continued)

C.1 and C.2

If a UHS train cannot be restored to OPERABLE status within the associated Completion Time, 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 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power.

SURVEILLANCE REQUIREMENTS

[The COL applicant to provide a surveillance for makeup water to UHS cooling tower.]

SR 3.7.19.1

This SR verifies that adequate short term (3 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the ESW pumps during the first 3 days post LOCA. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS basin water level is ≥ 23.75 feet from the bottom of the basin.

SR 3.7.19.2

This SR verifies that the UHS is available to cool the CCW System and EDG to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a postulated accident. With water temperature of the UHS basin ≤ 90°F, the design basis assumptions associated with initial UHS temperature are bounded. With the water temperature of the UHS basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Emergency Diesel Generators (EDG) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.19.3

Operating each cooling tower fan for ≥ 15 minutes in each speed setting and direction, including reverse verifies that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.

SR 3.7.19.4

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.19.5

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

[SR 3.7.19.6

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified emergency makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. The Frequency is in accordance with the Inservice Testing Program and is in accordance with the ASME OM Code (Ref. 5). This SR verifies that the UHS emergency makeup flowrate is $\geq$ 300 gpm.]

REFERENCES

1. FSAR Section 9.2.5.

2. FSAR Section 6.2.

3. FSAR Section 5.4.7.

4. Regulatory Guide 1.27, Rev. 2, January 1976.

[5. ASME Code for Operation and Maintenance of Nuclear Power Plants.]

B 3.7 PLANT SYSTEMS

B 3.7.20 Main Steam Line Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 4 of Appendix A to 10 CFR 50 (Ref. 1) requires components be designed to withstand the environmental and dynamic effects associated with both normal plant operation and postulated accidents. The U.S. EPR design applies the leak-before-break (LBB) methodology to eliminate the dynamic effects of main steam line pipe rupture from the steam generators to the first anchor point location (i.e., the Reactor Containment Building penetration) inside the Reactor Containment Building (Ref. 2). The analysis demonstrates that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping.

Leakage detection systems must have the capability to detect pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of leakage.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Humidity levels of the containment atmosphere as an indicator of potential leakage. Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump and condensate flow from air coolers. Humidity level monitoring is considered most useful as an direct alarm or indication to alert the operator to a potential problem.

The primary method used to detect leakage from the main steam line is the local humidity detection system, which has the capability of detecting a leakage of 0.1 gpm within four hours. Regulatory Guide 1.45 (Ref. 3) specifies a time frame of one hour for leakage detection. However, as noted in NUREG-1793 (Ref. 4), leakage detection for LBB purposes does not require the same degree of timeliness. A secondary method of detecting a leakage of 0.1 gpm within four hours for the main steam line is the containment sump level monitor.

BASES

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

Main steam line leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

One method of protecting against large main steam leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires the main steam local humidity detection system to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when main steam line leakage indicates possible main steam line degradation.

APPLICABILITY

Because of elevated steam generator temperature and pressure in MODES 1, 2, 3, and 4 when a steam generator is relied upon for heat removal, main steam line leakage detection instrumentation is required to be OPERABLE.

In MODES 4 when a steam generator is not relied upon for heat removal, 5, and 6, main steam line leakage detection instrumentation is not required because the Main Steam System pressure is greatly reduced from normal operating pressure, resulting in lower stresses and a reduced potential for leakage. In addition, the steam generators are not the primary method of RCS heat removal in MODES 5 and 6.

ACTIONS

A.1

With the main steam local humidity detection system inoperable, the containment sump level monitor can provide the equivalent information.

Restoration of the main steam local humidity detection system to OPERABLE status within a Completion Time of 30 days is required to regain the function after the system’s failure. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

BASES

ACTIONS (continued)

B.1

With the containment sump level monitor inoperable, the main steam local humidity detection system can provide the equivalent information.

Restoration of the containment sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the system's failure. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.20.1

A CHANNEL CHECK of the main steam local humidity detection system provides reasonable confidence that the instrument is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.7.20.2, SR 3.7.20.3

These SRs require the performance of a CALIBRATION for each of the main steam line leakage detection instrumentation channels. The CALIBRATION verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven that this Frequency is acceptable.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4.

2. FSAR Section 3.6.3.

3. Regulatory Guide 1.45, Rev. 1, May 2008.

4. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” U.S. Nuclear Regulatory Commission, September 2004.