B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND	The MSSVs provide overpressure protection for the secondary system. The MSSVs 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. Two MSSVs are located on each Main Steam Line, outside containment, upstream of the main steam isolation valves and downstream of the Main Steam Relief Train (MSRT), 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). The MSSV setpoints and capacities are such that with consideration of reactor trip, the MSSVs alone will prevent main steam pressure from rising above 110% of the steam generator design pressure upon full loss of load.
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 postulated accident. 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 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 over pressure. The safety analysis demonstrates that the transient response for MSSV closure occurring from full power without a direct reactor trip 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 MSSV closure event.

The safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE.

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(d)(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 102% 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.

APPLICABILITY

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

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

BASES

ACTIONS

A.1 and A.2

With one required MSSV inoperable, the associated MSRT is verified OPERABLE and action must be taken to restore the valve to OPERABLE status within 30 days. 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 C must be immediately entered.

The 30 day Completion Time considers the following:

a. With one MSSV inoperable, the resulting relief capacity of the affected SG is 75% (taking into account the MSRT) of the full load steam generation of the assigned steam generator, which is greater than the 50% relief capacity considered in the safety analysis.

b. The remaining OPERABLE overpressure protection devices are sufficient for heat removal for the long term phase.

The 30 day Completion Time is considered reasonable for restoring the inoperable components to OPERABLE status.

B.1

With two MSSVs inoperable, actions must be taken to restore the inoperable MSSVs to OPERABLE status within 7 days.

This Completion Time is applicable because:

a. With two MSSVs inoperable on the same SG, the resulting relief capacity of the affected SG is 50% (taking into account the MSRT) of the full load steam generation per SG, which is equal to 50% relief capacity considered in the safety analysis.

b. With one MSSV inoperable on one SG and one MSSV inoperable on another SG, the resulting relief capacity for each SG is 75% of the full load steam generation per SG. This combination of inoperabilities is different from the one in the safety analysis. However, the total relief capacity of the four SGs is 350% of the full load steam generation per SG, which is the exact relief capacity considered in the safety analysis.

c. The remaining OPERABLE overpressure protection devices (MSRTs) are sufficient for heat removal in the long term phase.

BASES

ACTIONS (continued)

The 7 day Completion Time is considered reasonable based on operating experience to accomplish the Required Action in an orderly manner without challenging unit systems.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met or if three 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 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

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 Code (Ref. 4) requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5).

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME 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, Class 2 Components.

3. FSAR Section 15.2.

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

5. ANSI/ASME OM-1-1987.

6. FSAR Section 15.4.

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 manifolds 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 Protection 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).

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).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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 cooldown. With a loss of offsite power, the response of mitigating systems is delayed. The worse case single failure is a main steam relief control valve associated with one of the unaffected steam generators failed in the fully open position (Ref. 3).

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.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

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

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

d. 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(d)(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.

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.

BASES

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 unit 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 unit systems.

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.

BASES

ACTIONS (continued)

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 unit systems.

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 Code (Ref. 5). This SR is modified by a Note that limits this surveillance to MODES 1 and 2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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.

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. 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 prior to entry into MODE 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), FW Low Load Control Valves (MFWLLCVs), MFW Very Low Load Control Valves (MFWVLLCVs), and MFW Main Isolation Valves (MFWMIVs)) are located in valve stations, physically separated from each other and from other systems. Within these valve compartments, the MFW lines are arranged in three trains, one Very Low Load, one Low Load, and one Full Load train. 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 MFWMIV. The very low load flow path for each steam generator includes one MFWVLLCV, one MFWLLIV, and the MFWMIV. Each of these trains can be isolated redundantly by one isolation valve, one control valve, or the MFWMIV. The Low Load isolation valve allows isolation of the Low Load and the Very Low Load train 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.

A MFW Isolation valve outside containment and a MFW check valve inside containment provide the containment isolation function.

The MFWFLIVs and MFWFLCVs close on a reactor trip. The low and low-low range control and isolation valves close in response to steam generator level as described in Reference 1. The MFWMIV closes on a containment isolation signal. 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 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 detail in Ref. 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(d)(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 SLB, or an SGTR. It also ensures that the MFWMIV provides isolation for events requiring containment isolation. This LCO requires that four MFWFLIVs, four MFWFLCVs, four MFWLLIVs, four MFWLLCVs, four MFWVLLCVs, and four MWFMIVs be OPERABLE. The MFWFLIVs, MFWFLCVs, MFWLLIVs, MFWLLCVs, MFWVLLCVs, and MFWMIVs 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.
APPLICABILITY	The feedwater 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.

BASES

APPLICABILITY (continued)

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 flow path.

A.1

With one valve in the full load flow path inoperable, action must be taken to restore the affected valve to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 7 day Completion Time is reasonable, based on operating experience.

B.1

With two valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the associated MFWFLCV, MFWFLIV, and MFWMIV; and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 72 hour Completion Time is reasonable, based on operating experience.

C.1

With three valves in the full load flow path inoperable, action must be taken to restore the affected valves to OPERABLE status within 8 hours. The 8 hour Completion Time takes into account the redundancy afforded by the redundant actuation trains on MFW full load flow path valves and the low probability of an event occurring during this time period that would require isolation of the MFW full load flow path. The 8 hour Completion Time is reasonable based on operating experience.

BASES

ACTIONS (continued)

D.1 and D.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 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.

E.1 and E.2

If the MFWs 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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFWFLIV, MFWFLCV, MFWLLIV, MFWLLCV, MFWVLLCV, and MFWMIV 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 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 a method for cooling the unit to residual heat removal (RHR) 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. The MSRT valves also provide secondary overpressure protection.

One MSRT is provided for each steam generator, outside containment, upstream of the Main Steam Safety Valves and the main steam isolation valves. Each MSRT consists of one main steam relief control valve (MSRCV) located downstream of one main steam relief isolation valve (MSRIV).

The main steam relief control valves are motorized control valves, normally open, which allow control of the steam generator steam pressure, and consequently control of the cooldown rate. 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 main steam relief isolation valves 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 close (fail safe position) on loss of power supply or on loss of Instrumentation and Control. The pilot valves must be energized to open the associated MSRIV.

The MSRIVs are normally closed, with the pilot valves kept closed (de-energized). The valves open automatically and quickly on demand from the Protection System.

A description of the main steam relief control valves and of the main steam relief isolation valves is found in FSAR Section 10.3 (Ref. 1)

Each MSRT minimum required capacity is 50% of the full steam generation of the assigned steam generator (for a design core power level of 4590 MWth), at a design pressure of 1,435 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 SG.

BASES

BACKGROUND (continued)

Each MSRT maximum capacity is limited to 50% of the full load steam generation of its assigned steam generator (at a design core power of 4590 MWth), at design pressure of 1,435 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 Protection System, but can be controlled manually by the operator.

APPLICABLE SAFETY ANALYSES

The design basis of the main steam relief valves is established by the capability to remove residual heat in a controlled manner and to cool down the plant to RHRS entry conditions at various rates (normal cooldown at 90°F/h, partial cooldown at 180°F/h). The design rate of partial cooldown is applicable to events with two steam generators OPERABLE, each with one MSRT OPERABLE.

The MSRTs residual heat removal function in the safety analysis of Reference 3 is required:

a. To perform residual heat removal at the controlled rate following Condition II, III, and IV events with the condenser inoperable;

b. To perform cooldown to RHR entry conditions, following Condition II, III, and IV events with the condenser inoperable; and

c. To perform Partial Cooldown of the unit at a rate of 180°F/h from MODE 3 to 870 psia to allow Medium Head Safety Injection (MHSI) into the Reactor Coolant System in the event of a Loss of Coolant Accident (LOCA) or Steam Generator Tube Rupture (SGTR).

The Main Steam Relief Trains do not directly participate in the reactivity control function. Nevertheless, reactivity control is supported by isolating a spuriously open MSRT to limit RCS cooling. Excessive increase in steam flow causes overcooling of the reactor coolant and thus reactivity feedback to the core.

In case of a SGTR, the MSRT participates in the confinement of radioactive material. In the SGTR mitigation process, an increase in the MSRV setpoint of the affected SG over the MHSI delivery pressure enables termination of the leak flow. It also prevents overfilling of the affected SG. In the event that the condenser in inoperable, the MSRT challenge avoids response of the associated MSSVs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MSRT participates in the limitation of SG pressure increase in the most limiting Category II overpressure transient, inadvertent closure of one MSIV, thus limiting the SG pressure peak to less than 110% of the design pressure, without MSSV challenge. In other overpressure transients (Category III and IV), the MSRT participates in the limitation of the pressure peak in conjunction with the associated MSSV (see B 3.7.1). For the most limiting overpressure transient of Category 4 (i.e. loss of secondary side heat sink at full power without reactor trip), the inoperability of one MSRT is considered in the safety analysis and all other overpressure protection devices (other MSRTs and all MSSVs) are challenged and thus must be operable.

In all analyzed events, two steam generators and consequently both their MSRTs are required for residual heat removal and plant cooldown, considering a single failure of one MSRT and preventive maintenance performed on either electrical division or emergency diesel at the moment of the accident with assumed Loss of Offsite Power.

In events analyzed in Reference 3, the MSRT ensures residual heat removal by performing either Partial Cooldown or Fast Cooldown, either by automatic or manual action, depending on the event.

The MSRV position is automatically controlled by the Protection System as a function of power level, provided that the MSRIV is closed. The MSRV position is such that:

a. Consequences of a spurious MSRT event are limited with regards to the Reactor Coolant System; and

b. Mitigation of overpressure transients is ensured.

If the MSRIV opens, the MSRV is automatically switched into SG pressure control mode.

The MSRT valves satisfy Criterion 3 of 10 CFR 50.36(d)(2)(ii).

LCO

Four main steam relief trains (MSRIVs and MSRCVs) are required to be OPERABLE so as to ensure residual heat removal by a minimum of two steam generators even in the case of preventative maintenance and a single failure affecting the other two steam generators and connected cooling systems (e.g., Emergency Feedwater System, MSRT).

Isolation capability is also required on the four MSRTs, since any steam generator can be affected by a spuriously opened MSRIV or by an SGTR.

BASES

LCO (continued)

Failure to meet the LCO either results in the inability to perform residual heat removal and plant cooldown following an event with an inoperable condenser, or in the inability to isolate a SG affected by an SGTR or a spurious MSRIV opening.

A main steam relief control valve is considered OPERABLE when it is capable of full opening and closing and when it is capable of providing controlled relief of the main steam flow, with support of related I&C systems.

A main steam relief isolation valve is considered OPERABLE when it is capable of opening and when it is capable of re-closure after challenge.

APPLICABILITY

In MODES 1, 2, and 3, the MSRTs are required to be OPERABLE to provide a decay heat removal path in conjunction with the Emergency Feedwater System.

In MODE 4, 5, or 6, decay heat removal is provided by the Low Head Safety Injection System.

ACTIONS

A.1

With one control line inoperable for opening in one or more MSRIVs (i.e. one pilot valve is blocked closed), the affected MSRIVs are still OPERABLE, however the control line(s) must be restored to OPERABLE status in 30 days. 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)

In case one pilot valve is blocked open in at least one control line 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.

B.1 and B.2

With one or two MSRIVs inoperable for opening (e.g., due to a mechanical failure or due to two pilots in parallel blocked closed), the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. The associated MSSV must be verified OPERABLE and the valves must be restored to OPERABLE status in 7 days. Verification of MSSV OPERABILITY is performed as an administrative check by examining logs or other information to determine if an MSSV is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the MSSV. If the OPERABILITY of the associated MSSV cannot be verified, however, Condition C must be immediately entered.

The Completion Time of 7 days is reasonable because:

a. In case of an event with loss of condenser, the two OPERABLE MSRTs are capable of performing the residual heat removal function. However the single failure criterion may not be met.

b. In case of an overpressure event, the redundancy provided by the two OPERABLE MSSVs ensure the limitation of SG pressure (the necessary relief capacity being 50% of full load steam generation per SG). However the single failure criterion is not met.

With one or two MSRCVs inoperable, the residual heat removal function and the overpressure protection of the corresponding MSRT are not assured, as well as the isolation function. The single failure criterion is not fulfilled any more, so the same Completion time of 7 days applies to restore MSRCV(s) to OPERABLE status.

Finally, with one or two MSRIVs inoperable for closing (e.g., blocked open during residual heat removal with MSRTs or due to a failed test), the residual heat removal function is still ensured, but the redundancy for MSRT isolation is lost because it can only be ensured by associated MSRVs. As a result, the same Completion Time of 7 days applies to restore to OPERABLE status.

BASES

ACTIONS (continued)

C.1 and C.2

If required Action A.1 or B.1 cannot be met within the required Completion Times, the unit must be placed in a MODE in which the LCO does not apply, and in which the inoperable MSRCV or MSRIVs can be restored to OPERABLE status. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

With three or more MSRIVs inoperable for opening, or three or more MSRCVs inoperable, the residual heat removal function and the overpressure protection function of the corresponding MSRT are not assured. Only one MSRT remains OPERABLE for this function, which is less than the needed two MSRTs for residual heat removal following analyzed events.

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 steam generators for heat removal, within 12 hours.

The MSRCVs inoperability also affects the MSRT isolation function by loss of redundancy (only MSRIVs can ensure the function).

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies each MSRIV OPERABLITY by opening the valve and then by closing the MSRIV. This SR is performed once every refueling outage on a STAGGERED TEST BASIS for each control line (i.e. twice per MSRIV) in hot shutdown conditions. The frequency is reasonable based on the fact that complete opening of an MSRIV is not possible during power operation and on the operating experience of similar MSRIVs on existing plants.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 Protection 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.

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 Protection System during power operation, which ensures that the valve is not blocked in a specific position.

REFERENCES

1. FSAR Section 10.3.

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

3. 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 on low Steam Generator (SG) level or Loss of Offsite Power. The EFW pumps take suction through a common supply header from their respective EFW storage pool (SP) 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 valves (MSRVs) (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 common supply header.

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) 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 main steam relief valves.

The EFW System actuates automatically on low steam generator water level signal generated by the Protection System (LCO 3.3.1). The system also actuates on loss of offsite power signal.

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.

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 steam generator safety valve 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 common 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 train whose pump is unavailable due to maintenance.

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

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

The Protection 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.

The EFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(d)(2)(ii) for operation in MODES 1, 2, and 3 and Criterion 4 of 10 CFR 50.36(d)(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 common 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 common supply and discharge headers. The EFW System is considered OPERABLE when the components and common 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 with only one EFW pump OPERABLE, operation is allowed to continue because only one EFW pump is 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, one EFW pump is sufficient to remove decay heat. Although not required, the unit may continue to cool down to LHSI entry conditions.

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE in the event that it is called upon to function when MFW and 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.

In MODE 4 and 5, the EFW System may be used for heat removal via the steam generators.

In MODE 6, the steam generators are not normally used for heat removal, and the EFW System is not required

ACTIONS

A Note prohibits the application of LCO 3.0.4.b for two or more EFW trains inoperable when entering MODE 1. There is an increased risk associated with entering MODE 1 with two or more EFW 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.

BASES

ACTIONS (continued)

A.1

With one EFW 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 train is not available due to maintenance, and the low probability of a postulated accident occurring during this time period.

B.1

With two EFW trains inoperable in MODES 1, 2, or 3, action must be taken to restore at least one inoperable EFW 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 and C.2

When any Required Action and associated Completion Time cannot be met; or if three EFW trains are inoperable in MODE 1, 2, or 3; or the common injection header; 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 within 12 hours.

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

D.1

With four EFW 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 train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one EFW 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.

BASES

ACTIONS (continued)

E.1

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 the one required EFW pump inoperable, action must be taken to immediately restore an inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW System flow paths 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. For the EFW System, this SR includes the steam generator blowdown isolation 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 3.7.5.2

Each EFW pump suction and supply header isolation valve is required to be locked open at 31 day intervals. This surveillance is designed to ensure that all EFW pumps can the inventory of all EFW pools.

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

SR 3.7.5.3

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

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 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 Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

SR 3.7.5.5

This SR verifies that EFW can be delivered to the appropriate steam generators in the event of any accident or transient that generates a Protection System actuation, by demonstrating that each automatic valve in the flow path actuates to its correct position, each EFW pump starts automatically, and flow rate is controlled within required limits and steam generator level is controlled within limits, 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 3.7.5.6

This SR verifies that the EFW is properly aligned by verifying the flow paths from the supply header 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.

BASES

REFERENCES

1. FSAR Section 10.4.9.

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

3. FSAR Section 15.2

B 3.7 PLANT SYSTEMS

B 3.7.6 Emergency Feedwater (EFW) Storage Pools

BASES

BACKGROUND	The EFW pumps take suction through separate suction lines from their respective EFW storage pool (SP) 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 steam 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 common supply header. Because the SPs are principal components in removing residual heat from the Reactor Coolant System (RCS), they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The SPs are designed to Seismic Category I to ensure availability of the feedwater supply. A description of the SPs is found in FSAR Section 10.4.9 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The EFW SPs provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in Chapters 6 and 15 (Ref. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally four hours at MODE 3, steaming through the MSSVs and MSRVs followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate or a lower cooldown rate if offsite power is not available. The limiting accident for the EFW SPs is a Main Feedwater Line Break (MFWLB) with a natural circulation cooldown. The EFW SPs satisfy the requirements of Criterion 2 and 3 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

To satisfy accident analysis assumptions, the EFW SPs must contain sufficient water to remove decay heat for four hours following a reactor trip from 102% RTP and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the EFW pumps during cooldown or before isolating EFW to a faulted steam generator.

The EFW SP required usable volume of 300,000 gallons is based on a cooldown to RHR entry conditions at 50°F/hour, with all four reactor coolant pumps in service. 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 the common supply header.

APPLICABILITY

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

In MODE 5 or 6, the EFW SPs are not required because the EFW System is not required.

ACTIONS

A.1 and A.2

With one EFW SPs inoperable in MODE 1, 2, or 3, or MODE 4, when a steam generator is being relied upon for heat removal, action must be taken to verify the usable volume in the remaining SPs is ≥ 300,000 gal. and to declare the associated EFW train inoperable.

B.1 and B.2

With two or more EFW SPs inoperable or the usable volume of the available SPs is < 300,000 gal., 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 on 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 unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the EFW Storage Pools 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 SPs have no other function that 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 SP levels.

SR 3.7.6.2

This SR verifies every 31 days that the EFW supply cross connect valves are locked open. This verification ensures that the usable volume in the SPs are available to all EFW trains through the supply cross connect header and ensures timely discovery if a valve should be not locked open. If an EFW supply cross connect valve is not open, the usable volume of the SP is not available to each of the four EFW trains as assumed in the safety analysis. 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 Chapter 6.

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 Protection 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.

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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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_{cold} < 350°F), to MODE 5 (T_{cold} < 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_{cold} < 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 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(d)(2)(ii).

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.

BASES

APPLICABILITY (continued)

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 indicated 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.

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.

BASES

ACTIONS (continued)

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 unit systems.

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, other than the RCP thermal barrier cooling common loop, 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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 3.7.7.3

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 one mechanical draft cooling tower, associated basin, pump, piping, valving, instrumentation, and mechanical filtration. Each safety related 2-cell seismic Category I mechanical draft cooling tower rejects energy from the ESW fluid to the ambient and returns the cooled fluid to the ESW cooling tower basin, from which the ESW pumps take suction. Each ESW 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 system pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA or loss of offsite power. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions.

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 1 emergency makeup water supply, to the ESW cooling tower basins, necessary to support 30 days of post accident mitigation is provided by the safety-related Ultimate Heat Sink (UHS) Makeup Water System that draws water from the Chesapeake Bay. Chesapeake Bay water enters the UHS Makeup Water Intake Structure through an intake channel shared the Circulating Water System Makeup Intake Structure. The UHS Makeup Water Intake Structure houses four independent UHS Makeup Water System trains, one for each ESW division. Each train has one pump, a discharge check valve, and a pump

BASES

BACKGROUND (continued)

discharge isolation motor operated valve, all housed in the UHS Makeup Water Intake Structure, plus the buried piping running up to and into the ESW pumphouse at the ESW cooling tower basin. Each UHS Makeup Water System pump is rated at 750 gpm.

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).

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.

Each ESW basin is sized for 3 days of post LOCA operation without requiring makeup. ESW 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 ESW 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 ESW basin, or equivalent make-up.

The ESW System satisfies Criterion 2 and 3 of 10 CFR 50.36(d)(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 two cooling tower fans, pump, associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE and the ESW basin contains ≥ 27.2 feet of water at ≤ 90°F with capability from makeup from OPERABLE source. An OPERABLE emergency makeup water source consists of one OPERABLE train of the UHS Makeup Water System capable of providing makeup water to its associated ESW cooling tower basin. Each UHS Makeup Water System train includes a pump, valves, piping, instruments and controls to ensure the transfer of the required supply of water from the Chesapeake Bay to its associated ESW cooling tower.

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 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 ESW basin water level is ≥ 27.2 feet from the bottom of the basin.

SR 3.7.8.2

This SR verifies that the ESW System 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 ESW basin ≤ 90°F, the design basis assumption associated with initial ESW temperature are bounded. With the water temperature of the ESW basin > 90°F, long term cooling capability of the Emergency Core Cooling System (ECCS) loads and Diesel Generators (DGs) may be affected. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.8.3

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.4

Operating each cooling tower fan for ≥ 15 minutes in all speed settings 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 ESW cooling tower fans occurring between surveillances.

SR 3.7.8.5

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.

SR 3.7.8.6

This SR verifies proper automatic operation of the ESW pumps and cooling tower fans 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.7

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified makeup flowrate ensures that sufficient NPSH can be maintained to operate the ESW pumps following the first 3 days post LOCA. 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. This SR verifies that the ESW makeup flowrate is ≥ 300 gpm.

REFERENCES

1. FSAR Section 9.2.1.
2. FSAR Section 6.2.
3. FSAR Section 5.4.7.
4. Regulatory Guide 1.27.

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 independent trains. Each train consists of a chiller refrigeration unit (three 50% compressors per unit), chilled water pumps (two 100% pumps), surge tank, piping, valving, and instrumentation. 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.

The SCW System is normally operating and cools the Control Room Air Conditioning System (CRACS), Safeguards Building Ventilation System Electrical Division (SBVSED), and the train 1 and 4 Low Head Safety Injection (LHSI) pump motor and seal coolers. The combined HVAC function of the SBVSED and SCW systems is backed by a non-safety related, 100% capacity maintenance train which is cooled by the Operational Chilled Water System.

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.

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 CRAC and SBVSED safety-related HVAC Systems in addition to the LHSI pump motor and seal coolers (train 1 and 4 only). 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 System, 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 the electrical rooms requires the availability of each train of SCW in order to ensure the ability of the plant to meet all required safety related functions during any AOO or postulated accident.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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(d)(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 post accident heat loads.

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

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 to OPERABLE status within 72 hours. In this condition, the three remaining OPERABLE SCW trains are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in one of the OPERABLE ESW train could result in loss of SCW System function.

The 72 hour 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.

BASES

ACTIONS (continued)

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 unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR requires verification every 24 hours that each SCW train is in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 24 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor SCW train performance.

SR 3.7.9.2

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.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the 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 SCW system is slow and is not expected over this time period.

SR 3.7.9.4

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.

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 or smoke.

The CREF consists of two 100% capacity iodine filtration trains which operate when radioactive contamination is detected at the site or inside the control room envelope (CRE) area. The iodine filtration train is a bypass path of the fresh air intake train for the Control Room Air Conditioning System (CRACS) normal air supply. The air from CRE can also be recirculated through the CREF Iodine Filtration trains. The iodine filtration trains are provided as bypass lines on two of the four normal CRACS air intake trains; other two CRACS intake trains do not have the bypass iodine filtration trains. During an emergency, the fresh outside air and recirculated air are directed through air intake motorized damper and electric heater through the CREF Iodine Filtration train. Each iodine filtration train consists of motorized damper, electric heater, prefilter, upstream HEPA filter, an activated carbon iodine filter, downstream HEPA filter, booster fan, and manual isolation damper. The filtered and clean air is then directed through one or both CRACS normal 75% capacity air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. The conditioned and clean air is then supplied to the CRE areas. Electric heaters are installed in the CRE supply air ducts to maintain individual room temperatures and relative humidity. The exhaust air from the CRE areas is directed through the recirculation air shaft and then recycled either through the iodine filtration trains or CRACS air conditioning trains. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers. The HEPA filter bank downstream of the carbon iodine filter collects carbon fines and provides backup in case of failure of the upstream HEPA filter bank. Continuous operation of each train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

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 during 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 inleakage of unfiltered air into the CRE will not exceed the inleakage 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.

The CREF train is an emergency system, which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), the outside fresh air supply to the CRE is isolated, and the outside air is directed through the CREF train. The CRE ventilation air is recycled through the air conditioning filter trains and/or CREF train.

Actuation of the CREF places the system in the emergency radiation mode of operation. Actuation of the system to the emergency radiation 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 CREF trains. The emergency radiation state also maintains control room pressurization and filtered ventilation of the air supply to the CRE.

Outside makeup air is supplied through the iodine filtration train and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

The outside air entering the CRE is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required.

BASES

BACKGROUND (continued)

One CREF operating at a flow rate of < 4000 cfm 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 one of the other filter trains. Normally open isolation dampers are arranged in series so the failure of one damper to shut will not result in a breach of isolation. The CREF train components are 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 postulated accident without exceeding a 5 rem whole body dose or its equivalent to any part of the body 5 rem total effective does 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 Chapter 15 (Ref. 2).

The CREF consists of two 100% capacity iodine filtration trains. Each iodine filtration train can be aligned with one of the two 75% capacity air conditioning trains. There are only two iodine filtration trains since only slow failure modes are assumed and filtration efficiency is checked periodically. Both CREF trains with the associated air conditioning 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.

The CREF provides protection from smoke to the CRE occupants. Reference 3 discusses that the need for protection of CRE occupants following a hazardous chemical release is not required at CCNPP Unit 3. 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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(d)(2)(ii).

LCO

In the event of a postulated accident, one iodine filtration 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 iodine filtration 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 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 smoke.

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

.

BASES

LCO (continued)

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.

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.

ACTIONS

A.1

With one CREF train inoperable, for reasons other than an inoperable CRE boundary, 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 trains to provide the required capability.

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

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 smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 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 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

BASES

ACTIONS (continued)

consequences, and that CRE occupants are protected from 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 90 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 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

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.

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.

BASES

ACTIONS (continued)

E.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREF trains 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 Iodine Filtration trains and associated Air Conditioning trains inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., Condition B), 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 from humidity in the ambient air should be performed. Each Iodine filtration 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.

SR 3.7.10.2

This SR verifies that the required CREF train 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 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. Chapter 15.
3. FSAR Section 6.4.
4. FSAR Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
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).

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 four normal system 75% capacity identical fresh air intake trains. For each intake train, the fresh air is taken from outside through motorized damper, electric heater, and prefilter. The fresh filtered air is then mixed with the CRE recycled air. The mixed air is then directed through one of the four 75% capacity associated air conditioning train. Each air conditioning train consists of volume control manual damper, cooling coil, moisture separator, fan suction and discharge silencers, supply air fan, HEPA filter, steam humidifier, non-return damper, volume control electric damper, and fire dampers. 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 temperatures and relative humidity. The exhaust air from the control room envelope (CRE) areas is directed through the recirculation air shaft and then recycled through the air conditioning trains upstream of the cooling coils for each train. The exhaust air from the CRE can also be recycled through the CREF Iodine Filtration trains if contamination is detected in the CRE. The exhaust from kitchen and sanitary areas is separated from the recycle return air and processed separately.

Two out of four 75% 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 65°F to 75°F, and humidity 40% to 60%.

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 65°F to 75°F. Redundant detectors and controls are provided for

BASES

APPLICABLE SAFETY ANALYSES (continued)

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(d)(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 three trains. These components include the heating and cooling coils, moisture separators, humidifiers, and associated temperature control instrumentation. In addition, the CRACS must be operable to the extent that air circulation can be maintained.

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 or two CRACS train(s) inoperable, action must be taken to restore 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 the OPERABLE CRACS train 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.

BASES

ACTIONS (continued)

B.1 and B.2

If any Required Action and Associated Completion Time of Condition A 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 unit systems.

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CRACS train cannot be restored to OPERABLE status within the required Completion Time, the 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 C.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.

D.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.

E.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 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 buildings 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 plant 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, electric heater, pre-filter, 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 plant stack. The downstream 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 pre-filters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and carbon adsorbers.

BASES

BACKGROUND (continued)

In case of a LOCA with assumed ECCS leakage, the accident air exhaust from the safeguard buildings and fuel building is also directed through the accident iodine exhaust filtration trains prior to release through the plant 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 buildings 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 buildings 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(d)(2)(ii).

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 buildings and fuel building exceeding the 10 CFR 50.34 (Ref. 6) limits in the event of a LOCA.

BASES

LCO (continued)

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 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 hot 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.

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.

BASES

ACTIONS (continued)

B.1

If the safeguard buildings 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 buildings and fuel building boundaries within 24 hours. During the period that the safeguard buildings or fuel building boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19 and 10 CFR Part 100 shall be utilized to protect plant personnel from potential hazards such as radioactive contamination,smoke, temperature and relative humidity, and physical security. Preplanned measures shall be available and implemented upon entry into the condition to address these concerns regardless of whether the entry is intentional or unintentional entry. 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 buildings or fuel building boundary.

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 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 unit systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Verifying that safeguards building 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 safeguards building and fuel building OPERABILITY requires verifying each access opening door is closed. However, all safeguards building 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.

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.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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.

SR 3.7.12.6 and 3.7.12.7

The SBVS exhausts the safeguards building and fuel building atmosphere to the environment through appropriate treatment equipment. Each safety SBVS train is designed to draw down the safeguards building and fuel building to a negative pressure ≥ 0.25 inches of water gauge (wg) in ≤ 305 seconds and maintain the safeguards building and fuel building at a negative pressure ≥ 0.25 inches wg at a flow rate ≤ 2,400 cfm from the safeguards building and fuel building. To ensure that all fission products released to the safeguards building and fuel building are treated, SR 3.7.12.6 and SR 3.7.12.7 verify that a pressure in the safeguards building and fuel building that is less than the lowest postulated pressure external to the safeguards building and fuel building boundaries can be established and maintained. When the SBVS is operating as designed, the establishment and maintenance of safeguards building and fuel building pressure cannot be accomplished if the safeguards building 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 safeguards building and fuel building can be maintained at a negative pressure ≥ 0.25 inches wg. The primary purpose of these SRs is to ensure safeguards building and fuel building boundary integrity. The secondary purpose of these SRs is to ensure that the SBVS train being tested functions as designed. There is a separate LCO with Surveillance Requirements which serves the primary purpose of ensuring OPERABILITY of the SBVS. 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 safeguards building and fuel building OPERABILITY. Operating experience has shown the safeguards building 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.

BASES

REFERENCES

1. FSAR Section 9.4.6.
2. FSAR Section 9.4.3.
3. FSAR Section 9.4.5.
4. FSAR Section 15.0.
5. Regulatory Guide 1.25.
6. 10 CFR 50.34.
7. Regulatory Guide 1.52, Rev. 3.

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(d)(2)(ii).
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.

BASES

LCO (continued)

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 unit 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 Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. 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(d)(2)(ii).
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.

BASES

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.

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 Storage Pool Boron Concentration

BASES

BACKGROUND	The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel 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 0.995, including uncertainties and tolerances. The NRC guidelines specify 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. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has a potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the location of each assembly in accordance with LCO 3.7.16, “Spent Fuel Pool Storage.” Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.
APPLICABLE SAFETY ANALYSES	Although credit for the soluble boron normally present in the spent fuel pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity even in the absence of soluble boron. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, boiling, assembly dropped on top of a rack, lateral rack module movement and misplacement of a spent fuel assembly have been analyzed. The spent fuel pool k_{eff} storage limit of 0.95 is maintained during these events by a minimum boron concentration of 500 ppm with boric acid enriched to ≥ 37% B^{10} established by criticality analysis (Ref. 2). Compliance with the LCO minimum boron concentration limit of 500 ppm with boric acid enriched to ≥ 37% B^{10} ensures that the credited concentration is always available. The concentration and enrichment of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

BASES

LCO

The spent fuel storage pool boron concentration is required to be ≥ 500 ppm boron enriched to ≥ 37% B^{10}. The specified concentration and enrichment of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 2. This concentration of dissolved boron is the minimum required concentration and enrichment for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

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

When the concentration or enrichment of boron in the spent fuel storage 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 storage pool fuel locations, to ensure proper locations of the fuel, 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.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. 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 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 fuel storage pool is within the required limit. As long as this SR and SR 3.7.15.2 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 within limit 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. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).

2. UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

BASES

BACKGROUND

The high density spent fuel storage racks are divided into two separate and distinct regions as shown in Figure 4.3-1. Region 1, with a maximum of 360 storage locations, is designed to accommodate new fuel assemblies with a maximum enrichment of 5.0 weight percent U-235, or spent fuel assemblies regardless of the combination of initial enrichment and burnup. Region 2, with a maximum of 1000 storage locations, is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup limits specified in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. For storage of fuel in the spent fuel 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 faction (k_{eff}) of the fuel assembly array will be less than 0.995, including uncertainties and tolerances. The NRC guidelines specify 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.

The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal and accident conditions, since only a single independent accident need be considered at one time. For example, the only accident scenario that has the potential for more than negligible positive reactivity effect is an inadvertent misplacement of a new fuel assembly. This accident has is a potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. To mitigate these postulated criticality related accidents, enriched boron is dissolved in the pool water. Safe operation with unborated water and no movement of assemblies may, therefore, be achieved by controlling the combination of initial enrichment and burnup of the stored fuel in accordance with the accompanying LCO. Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

BASES

APPLICABLE SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly (Refs. 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.15, "Spent Fuel Pool Boron Concentration") prevents criticality. By closely controlling the movement of each assembly and by checking the location of each 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 storage pool satisfies Criterion 2 of 10 CFR 50.36(d)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within Region 2 of the spent fuel pool in the accompanying LCO, ensure the k_{eff} of the spent fuel storage pool will always remain < 0.995, assuming the pool to be flooded with unborated water and < 0.95, with a boron concentration of greater than 500 ppm and boron enrichment ≥ 37%.

Storage of spent fuel is permitted in all Region 2 locations provided that the spent fuel meets the combination of initial enrichment and burnup requirements shown in Figure 3.7.16-1, Fuel Assembly Burnup Requirements for Region 2.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool.

ACTIONS

A.1

When the requirements of the LCO are not met, action must be immediately initiated to move the non-complying fuel assembly to an acceptable storage location (i.e., Region 1).

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.16-1 prior to storing the fuel assembly in Region 2.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2).

2. UN-TR-08-001, “Spent and New Fuel Storage Analyses for U.S. EPR Topical Report,” March 2008.

3. U.S. EPR FSAR Section 15.0.3.10, “Fuel Handling Accident.”

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 Low Head Safety Injection 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(d)(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 unit 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.