B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary 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 and Circulating Water System, is not available.

Eight MSSVs are located on each main steam header, outside the Containment Structure, upstream of the main steam isolation valves (MSIVs), as described in Reference 1, Chapter 10. The MSSV rated capacity passes the full steam flow at 102% RATED THERMAL POWER (100% + 2% for instrument error) with the valves full open. This meets the requirements of Reference 2, Section III, Article NC-7000, Class 2 Components. The MSSV design includes staggered setpoints, according to Table 3.7.1-1 in the accompanying Limiting Condition for Operation (LCO), so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering, because of insufficient steam pressure to fully open all valves, following a turbine reactor trip. The MSSVs have "R" size orifices.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2, Section III, Article NC-7000, Class 2 Components; their purpose is to limit secondary system pressure to $\leq$ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence or accident considered Reference 1, Chapter 14.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in Reference 1, Section 14.5. Of these, the full power loss of load event is the limiting anticipated operational occurrence. A loss of load isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Before delivery of auxiliary feedwater (AFW) to the steam generators, RCS pressure reaches peak pressure. The peak pressure is < 110% of the design pressure of 2500 psig, but high enough to actuate the pressurizer safety valves.

Although the Power Level-High Trip is not credited in the loss of load safety analysis, reducing the Power Level-High Trip setpoint ensures the Thermal Power limit supported by the safety analysis is met.

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

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, Section III, Article NC-7000, Class 2 Components, even though this is not a requirement of the Design Basis Accident (DBA) analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2, Section III, Article NC-7000, Class 2 Components requirements), and adjustment to the Reactor Protective System trip setpoints to meet the transient analysis limits. These limitations are according to those shown in Table 3.7.1-1, Required Action A.2, and Required Action A.3 in the accompanying LCO.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. An MSSV is considered inoperable if it fails to open upon demand.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

A Note is added to Table 3.7.1-2, stating that lift settings for a given steam line are also acceptable, if any two valves lift between 935 and 1005 psig, any two other valves lift between 935 and 1035 psig, and the four remaining valves lift between 935 and 1050 psig. Thus, the MSSVs still perform that design basis function properly.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the reactor coolant pressure boundary.

APPLICABILITY

In MODEs 1, 2, and 3, a minimum of five MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODEs.

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 MSSVs to be OPERABLE in these MODEs.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets Code requirements for the power level. The number of inoperable MSSVs will determine the necessary level of reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the power level-high channels. The setpoints in Table 3.7.1-1 have been verified by transient analyses.

The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.

The four-hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 hours is allowed in Required Action A.2 to reduce the setpoints. The Completion Time of 36 hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status in the associated Completion Time, or if one or more steam generators have less than five MSSVs OPERABLE, 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 Surveillance Requirement (SR) verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The safety and relief valve tests are to be performed in accordance with Reference 3. According to Reference 3, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/American Society of Mechanical Engineers (ASME) Standard requires that all valves be tested every five years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities, as found lift acceptance range, and frequencies necessary to satisfy the requirements. Table 3.7.1-2 defines the lift setting range for each MSSV for OPERABILITY; however, the valves are reset to ± 1% during the surveillance test to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This is to allow testing of the MSSVs at hot conditions. 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.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)
2. ASME, Boiler and Pressure Vessel Code
3. ANSI/ASME OM-1-1987, Code for the Operation and Maintenance of Nuclear Power Plants, 1987

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). Main steam isolation valve closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside, but close to, the Containment Structure. The MSIVs are downstream from the MSSVs, atmospheric dump valves (ADVs), and AFW pump turbine steam supplies to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a steam generator isolation signal generated by low steam generator pressure or on a containment spray actuation signal (CSAS) generated by high containment pressure. The MSIVs fail closed on loss of control or actuation power. The steam generator isolation signal also actuates the main feedwater isolation valves (MFIVs) to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in Reference 1, Section 10.1.

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside the Containment Structure, as discussed in Reference 1, Section 14.20. It is also influenced by the accident analysis of the SLB events presented in Reference 1, Section 14.14. The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for main SLB Containment Structure response is 75% power, no loss of offsite power, and failure of a steam generator feed pump to trip. This case results in continued feeding of the affected steam generator and maximizes the energy release into the Containment Structure.

This case does not assume failure of an MSIV; however, an important assumption is both MSIVs are OPERABLE. This prevents blowdown of both steam generators assuming failure of an MSIV to close.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside the Containment Structure upstream of the MSIV is the limiting SLB for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside the Containment Structure at hot full power is the limiting case for a post-trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.

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

a. An HELB inside the Containment Structure. In order to maximize the mass and energy release into the Containment Structure, the analysis assumes steam is discharged into the Containment Structure from both steam generators until closure of the MSIV occurs. After MSIV closure, steam is discharged into the Containment Structure only from the affected steam generator.

b. A break outside of the Containment Structure and upstream from the MSIVs. This scenario 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 limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This type of break will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves (e.g., excess load event) will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIV isolates the affected steam generator from the intact steam generator and minimizes radiological releases. The operator is then required to maintain the pressure of the steam generator with the ruptured tube below the MSSV setpoints, a necessary step toward isolating the flow through the rupture.

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

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

LCO

This LCO requires that the MSIV in each of the two steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, 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 as described in Reference 1, Chapter 14.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1 and in MODEs 2 and 3, except when all MSIVs are closed. In these MODEs there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing their safety function.

In MODE 4, the steam generator energy is low; therefore, the MSIVs are not required to be OPERABLE.

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

ACTIONS

<u>A.1</u>

With one MSIV inoperable in MODE 1, time is allowed to restore the component to OPERABLE status. Some repairs can be made to the MSIV with the unit hot. The eight hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

B.1

If the MSIV cannot be restored to OPERABLE status within eight hours, 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 six hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2, and close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C 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 eight hour Completion Time is consistent with that allowed in Condition A.

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, 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 seven day Completion Time is reasonable, based on engineering judgment, MSIV status indications available in the Control Room, and other administrative controls, to ensure these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or 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 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

BASES

MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is < 5.2 seconds. The MSIV closure time is assumed in the accident and containment analyses.

The Frequency for this SR is in accordance with the Inservice Testing Program. The MSIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the SR when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR

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

B 3.7 PLANT SYSTEMS

B 3.7.3 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the RCS upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the condensate storage tank (CST) (LCO 3.7.4) and pump to the steam generator secondary side via separate and independent connections, to the AFW header outside the Containment Structure. 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 MSSVs (LCO 3.7.1) or ADVs. If the main condenser is available, steam may be released via the steam bypass valves and the resulting excess water inventory in the hotwell is moved to the backup water supply.

The AFW System consists of, one motor-driven AFW pump and two steam turbine-driven pumps configured into two trains. The motor-driven pump provides 100% of AFW flow capacity; each turbine-driven pump can provide 100% of the required capacity to the steam generators as assumed in the accident analysis, but only one turbine-driven pump is lined up to auto start. The other turbine-driven pump is placed in standby and requires a manual start, when it is needed. The pumps are equipped with a common recirculation line to prevent pump operation against a closed system. The motor-driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

One pump at full flow is sufficient to remove decay heat and cool the unit to Shutdown Cooling (SDC) System entry conditions.

The steam turbine-driven AFW pumps receive steam from either main steam header upstream of the MSIV. Each of the steam feed lines will supply 100% of the requirements of the turbine-driven AFW pump. The turbine-driven AFW pump supplies a common header capable of feeding both steam generators, with air-operated valves (with controllers powered by AC vital buses) actuated to the appropriate steam generator by the Auxiliary Feedwater Actuation System (AFAS).

The AFW System may also supply feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions although the normal supply is main feedwater (MFW).

The AFW 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 AFW System supplies sufficient water to cool the unit to SDC entry conditions, and steam is released through the ADVs.

The AFW System actuates automatically on low steam generator level by the AFAS, as described in LCO 3.3.4. The AFAS logic is designed to feed either or both steam generators with low levels, but will isolate the AFW System from a steam generator having a significantly lower steam pressure than the other steam generator. The AFAS automatically actuates one AFW turbine-driven pump and associated air-operated valves (with controllers powered by AC vital buses) when required, to ensure an adequate feedwater supply to the steam generators. Air-operated valves with controllers powered by AC vital busses are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System is discussed in Reference 1.

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with a loss of normal feedwater.

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

The limiting DBAs and transients for the AFW System are as follows:

a. Main SLB; and

b. Loss of normal feedwater.

The AFW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform its design safety function. A train consists of one pump and the piping, valves, and controls in the direct flow path. Three AFW pumps are installed, consisting of one motor-driven and two non-condensing steam turbine-driven pumps. For a shutdown, only one pump is required to be operating, the others are in standby. Upon automatic initiation of AFW, one motor-driven and one turbine-driven pump automatically start.

The AFW System is considered to be OPERABLE when the components and flow paths required to provide AFW flow to the steam generators are OPERABLE. This requires that the motor-driven AFW pump be OPERABLE and capable of supplying AFW flow to both steam generators. The turbine-driven AFW pumps shall be OPERABLE with redundant steam supplies from each of the two main steam lines upstream of the MSIVs and capable of supplying AFW flow to both of the two steam generators. The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by a Note that allows AFW trains required for Operability to be taken out-of-service under administrative control for the performance of periodic testing. This LCO note allows a limited exception to the LCO requirement and allows this condition to exist without requiring any Technical Specification Condition to be entered. The following administrative controls are necessary during periodic testing to ensure the operator(s) can restore the AFW train(s) from the test configuration to its operational configuration when required. A dedicated operator(s) is stationed at the control station(s) with direct communication to the Control Room whenever the train(s) is in the testing configuration. Upon completion of the testing the trains are returned to proper status and verified in proper status by independent operator checks. The administrative controls include certain operator restoration actions that are virtually certain to be successful during accident conditions. These actions include but are not limited to the following: operation of pump discharge valves, operation of trip/throttle valve(s), simple handswitch/controller manipulations, and adjusting the local governor speed control knob. The administrative controls do not include actions to restore a tripped AFW pump due to the complicated nature of this task. Periodic tests include those tests that are performed in a controlled manner similar to surveillance tests, but not necessarily on the established surveillance test schedule, such as post-maintenance tests. This Note is necessary because of the AFW pump configuration.

APPLICABILITY

In MODEs 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory and maintain the RCS in MODE 3.

In MODE 4, the AFW System is not required, however, it may be used for heat removal via the steam generator although the preferred method is MFW.

In MODEs 5 and 6, the steam generators are not normally used for decay heat removal, and the AFW System is not required.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 and A.2

With one of the required steam-driven AFW pumps inoperable, action must be taken to align the remaining OPERABLE steam-driven pump to automatic initiating status. This Required Action ensures that a steam-driven AFW pump is available to automatically start, if required. If the OPERABLE AFW pump is properly aligned, the inoperable steam-driven AFW pump must be restored to OPERABLE status (and placed in either standby or automatic initiating status, depending upon whether the other steam-driven AFW pump is in standby or automatic initiating status) within seven days. The 72 hour and seven day Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and flow paths remain to supply feedwater to the steam generators.

B.1 and B.2

With the motor-driven AFW pump inoperable, action must be taken to align the standby steam-driven pump to automatic initiating status. This Required Action ensures that another AFW pump is available to automatically start, if required. If the standby steam-driven pump is properly aligned, the inoperable motor-driven AFW pump must be restored to OPERABLE status within seven days. The 72-hour and seven day, Completion Times are reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. Two AFW pumps and one flow path remain to supply feedwater to the steam generators.

C.1, C.2, C.3, and C.4

With two AFW pumps inoperable, action must be taken to align the remaining OPERABLE pump to automatic initiating status and to verify the other units motor-driven AFW pump is OPERABLE, along with an OPERABLE cross-tie valve, within one hour. If these Required Actions are completed within the Completion Time, one AFW pump must be restored to OPERABLE status within 72 hours. Verifying the other unit's motor-driven AFW pump is OPERABLE provides an additional level of assurance that AFW will be available if needed, because the other unit's AFW can be cross-connected if necessary. The cross-tie valve to the opposite unit is administratively verified OPERABLE by confirming that SR 3.7.3.2 has been performed within the specified Frequency. These one hour Completion Times are reasonable based on the low probability of a DBA occurring during the first hour and the need for AFW during the first hour. The 72 hour completion time to restore one AFW pump to OPERABLE status takes into account the cross-connected capability between units and the unlikelihood of an event occurring in the 72 hour period.

D.1

With one of the required AFW trains inoperable for reasons other than Condition A, B, or C (e.g., flowpath or steam supply valve), action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine-driven AFW pumps. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the AFW System, the time needed for repairs, and the low probability of a DBA event occurring during this period. One AFW train remains to supply feedwater to the steam generators.

E.1 and E.2

When the Required Action and associated Completion Time of Condition A, B, C, or D cannot be met the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 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.

F.1

Required Action F.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status.

With two AFW trains inoperable in MODEs 1, 2, and 3, the unit may be in a seriously degraded condition with only non-safety-related means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. However, a power change is not precluded if it is determined to be the most prudent action. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status. While other plant conditions may require entry into LCO 3.0.3, the ACTIONS required by LCO 3.0.3 do not have to be completed because they could force the unit into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the AFW water and steam supply flow paths, provides assurance that the proper flow paths exist for AFW 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 SR does not require any testing or valve manipulations; rather, it involves verification that those valves capable of potentially 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 3.7.3.2

Cycling each testable, remote-operated valve that is not in its operating position, provides assurance that the valves will perform as required. Operating position is the position that the valve is in during normal plant operation. This is accomplished by cycling each valve at least one cycle. This SR ensures that valves required to function during certain scenarios, will be capable of being properly positioned. The Frequency is based on engineering judgment that when cycled in accordance with the Inservice Testing Program, these valves can be placed in the desired position when required.

SR 3.7.3.3

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head (≥ 2800 ft for the steam-driven pump and ≥ 3100 ft for the motor-driven pump), ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Reference 2. Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing, discussed in Reference 2, at three month intervals satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred up to 24 hours until suitable test conditions are established. This deferral is required because there is an insufficient steam pressure to perform the test.

SR 3.7.3.4

This SR ensures that AFW can be delivered to the appropriate steam generator, in the event of any accident or transient that generates an AFAS signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal (verification of flow-modulating characteristics is not required). This SR 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 test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. The 24 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note indicating that the SR should be deferred up to 24 hours until suitable test conditions have been established.

SR 3.7.3.5

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month

Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note. The Note indicates that the SR should be deferred up to 24 hours until suitable test conditions are established.

SR 3.7.3.6

This SR ensures that the AFW system is capable of providing a minimum nominal flow to each flow leg. This ensures that the minimum required flow is capable of feeding each flow leg. The test may be performed on one flow leg at a time. The SR is modified by a Note which states, the SR is not required to be performed for the AFW train with the turbine-driven AFW pump until up to 24 hours after reaching 800 psig in the steam generators. The Note ensures that proper test conditions exist prior to performing the test using the turbine-driven AFW pumps. The 24 month Frequency coincides with performing the test during refueling outages.

SR 3.7.3.7

This SR ensures that the AFW System is properly aligned by verifying the flow path to each steam generator prior to entering MODE 2 operation, after 30 days in MODEs 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment, and other administrative controls to ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, the OPERABILITY of the flow paths is verified following extended outages to determine that no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. Minimum nominal flow to each flow leg is ensured by performance of SR 3.7.3.6.

REFERENCES

1. UFSAR, Section 10.3
2. ASME Code for Operation and Maintenance of Nuclear Power Plants

B 3.7 PLANT SYSTEMS

B 3.7.4 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the RCS. The CST provides a passive flow of water, by gravity, to the AFW System (LCO 3.7.3). The steam produced is released to the atmosphere by the MSSVs or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

The component required by this Specification is CST No. 12.

When the MSIVs are open, the preferred means of heat removal is to discharge steam to the condenser by the non-safety grade path of the turbine bypass valves. The condensed steam is returned to the backup water supply (CST No. 11 and CST No. 21) by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from an alternate source.

There is one CST (CST No. 12) shared by Units 1 and 2. A description of the CST is found in Reference 1, Sections 6.3.5.1 and 10.3.2.

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and to cool down the unit following all events except for the maximum hypothetical accident and the fuel handling accident in the accident analyses, discussed in Reference 1, Chapter 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the thermal analysis assumption is generally six hours at MODE 3, steaming through the ADVs and MSSVs followed by a cooldown to SDC entry conditions at the design cooldown rate. The dose analysis assumption is an eight hour cooldown to maximize Control Room and offsite doses.

The limiting event for the condensate volume is the large feedwater line break with a coincident loss of offsite power. Single failures that also affect this event include the following:

a. The failure of the diesel generator powering the motor-driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and

b. The failure of the steam driven train (requiring a longer time for cooldown using only one motor-driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

The CST satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

To satisfy accident analysis assumptions, CST No. 12 must contain sufficient cooling water for both units to ensure that sufficient water is available to maintain the RCS at MODE 3 for six hours following a reactor trip from 102% RATED THERMAL POWER, 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 AFW pumps during the cooldown while in MODE 3, as well as to account for any losses from the steam-driven AFW pump turbine, or before isolating AFW to a broken line.

The CST usable volume required is ≥ 150,000 gallons per unit (300,000 gallons for both units) in the MODE of Applicability. The 300,000 gallons of water is enough to provide for decay heat removal and cooldown of both units. By adjusting the feedwater flow to the permissible cooldown rate, decay heat removal and cooldown of both units can be accomplished in six hours. The 300,000 gallons are also adequate to maintain the RCS in MODE 3 for six hours with steam discharge to atmosphere with concurrent and total loss of offsite power, or to remove decay heat from both units for more than ten hours after initiation of cooldown and still maintain normal no-load water level in the steam generators. The total water volume in the tank includes the usable volume and water not usable because of the tank discharge line location.

OPERABILITY of the CST is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODEs 1, 2, and 3, the CST is required to be OPERABLE.

In MODEs 4, 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup water supply (CST No. 11 for Unit 1 and CST No. 21 for Unit 2) must be verified by administrative means within 4 hours and once every 12 hours thereafter.

OPERABILITY of the backup feedwater supply must include verification that the manual valves in the flow paths from the backup supply to the AFW pumps are open, and availability of the required volume of water (150,000 gallons) in the backup supply. The CST must be returned to OPERABLE status within seven days, as the backup supply may be performing this function in addition to its normal functions. The four hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The seven day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

If the CST volume is less than 300,000 gallons and greater than 150,000 gallons and both units are in the MODE of Applicability, only one unit must enter this condition provided the unit aligns to the OPERABLE backup water supply (CST No. 11 or CST No. 21).

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the affected unit(s) must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit(s) 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 plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CST contains the required usable volume of cooling water. (This volume $\geq$ 150,000 gallons per unit in the MODE of Applicability.) The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the Control Room, including alarms, to alert the operator to abnormal CST volume deviations.

Although the volume in the CST for each unit is required to be 150,000 gallons, the total combined volume for both units is 300,000 gallons.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.5 Component Cooling (CC) System

BASES

BACKGROUND

The CC System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, the CC System also provides this function for various nonessential components. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Saltwater (SW) System, and thus to the environment.

The CC System consists of two redundant loops that are always cross-connected. A loop consists of one of three redundant pumps, one of two redundant CC heat exchangers along with a common head tank, associated valves, piping, instrumentation, and controls. The third pump, which is an installed spare, can be powered from either electrical train. The redundant cooling capacity of this system, assuming single active failure, is consistent with the assumptions made in the accident analysis.

During normal operation one loop typically provides cooling water with a maximum CC heat exchanger outlet temperature of 95°F (a range of 70°F-95°F is acceptable during normal operating conditions) with the redundant loop components in standby. If needed, the redundant loop components can be aligned to supplement the in service loop. While operating on SDC with one loop, the CC heat exchanger outlet temperature may rise to a maximum temperature of 120°F.

Following a loss of coolant accident (LOCA) while recirculating water from the containment sump, the CC heat exchangers are designed to provide a maximum outlet cooling water temperature of 120°F provided one of the following component alignment combinations is met (assumes CC to containment and evaporators is isolated): a) 1 CC pump, 2 CC heat exchangers, and 2 SDC heat exchangers; b) 1 CC pumps, 1 CC heat exchanger, 1 SDC heat exchangers; and c) 2 CC pumps, 2 CC heat exchangers, 1 SDC heat exchangers. In the event of a passive failure of the common portions of the CC loop during a LOCA, the entire system would be lost. The unit can still be maintained in a safe condition since the containment coolers would be utilized in lieu of the spray pumps/shutdown heat exchangers to cool the Containment Structure (Reference 1, Section 9.5.5).

Additional information on the design and operation of the system, along with a list of the components served, is presented in Reference 1, Section 9.5.2.1. The principal safety-related function of the CC System is the removal of decay heat from the reactor via the SDC System heat exchanger. This may utilize the SDC heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a LOCA.

APPLICABLE SAFETY ANALYSES

The design basis of the CC System is for it to support a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 30 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the RCS by the safety injection pumps.

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

The CC System also functions to cool the unit from SDC entry conditions (T_{cold} < 300°F) to T_{cold} < 140°F during normal operations. The time required to cool from 300°F to 140°F is a function of the number of CC and SDC loops operating. One CC loop is sufficient to remove decay heat during subsequent operations with T_{cold} < 140°F. This assumes that a maximum inlet SW temperature occurs simultaneously with the maximum heat loads on the system.

The CC System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The CC loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CC loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CC loops must be OPERABLE. At least one CC loop will operate assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, the containment cooling function will also operate assuming the worst case passive failure post-recirculation actuation signal (RAS).

A CC loop is considered OPERABLE when the following:

a. The associated pump and common head tank are OPERABLE; and

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

The isolation of CC from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CC System.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CC System is a normally operating system that must be prepared to perform its post accident safety functions, primarily RCS heat removal by cooling the SDC heat exchanger.

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

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating the requirement of entry into the applicable Conditions and Required Actions of LCO 3.4.6, for SDC made inoperable by CC. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

With one CC loop inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC loop is adequate to perform the heat removal function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CC loop 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 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the CC flow path provides assurance that the proper flow paths exist for CC 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 SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

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

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

This SR verifies proper automatic operation of the CC valves on an actual or simulated safety injection actuation signal

(SIAS). The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This SR 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 test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.5.3

This SR verifies proper automatic operation of the CC pumps on an actual or simulated SIAS. The CC 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 test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.6 Service Water (SRW) System

BASES

BACKGROUND

The SRW System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation or a normal shutdown, the SRW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SRW System consists of two separate, 100% capacity safety-related cooling water subsystems. Each subsystem consists of a 100% capacity pump, head tank, two SRW heat exchangers, piping, valves, and instrumentation. A third pump, which is an installed spare, can be powered from either electrical train. The pumps and valves are remote manually aligned, except in the unlikely event of a LOCA. The pumps are automatically started upon receipt of a SIAS and all essential valves are aligned to their post-accident positions.

During normal operation, both subsystems are required, and are independent to the degree necessary to assure the safe operation and shutdown of the plant-assuming a single failure. During shutdown, operation of the SRW System is the same as normal operation, except that the heat loads are reduced. Additional information about the design and operation of the SRW System, along with a list of the components served, is presented in Reference 1, Section 9.5.2.2. In the event of a LOCA, the SRW System automatically realigns to isolate Turbine Building (non-safety-related) loads creating two independent and redundant safety-related subsystems. Service water flow to the spent fuel pool (SFP) cooler and the blowdown heat exchanger is automatically isolated as required for the DBA. Each SRW subsystem will supply cooling water to a diesel generator and two containment air coolers. However, the No. 11 SRW subsystem only supplies two containment air coolers since the No. 1A Diesel Generator is air cooled. Each SRW subsystem is sufficiently sized to remove the maximum amount of heat from the containment atmosphere while maintaining the SRW supply temperature to the diesel generator below its design limit.

APPLICABLE SAFETY ANALYSES

The design basis of the SRW System is for it to support a 100% capacity containment cooling system (containment coolers) and to remove core decay heat 30 minutes following a design basis LOCA, as discussed in Reference 1, Section 14.20. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the RCS by the safety injection pumps. The SRW System is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SRW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two SRW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming that worst case passive failure post-RAS.

An SRW subsystem is considered OPERABLE when:

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

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

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

With one SRW heat exchanger inoperable, action must be taken to restore operable status within 7 days. Isolating flow to one associated containment cooling unit will reduce the DBA heat load of the affected SRW subsystem to within the capacity of one SRW heat exchanger, thus ensuring that the SRW temperatures can be maintained within their design limits. This will allow the associated diesel generator (except for 11 SRW which does not cool a diesel generator) to remain operable. In this Condition, the other OPERABLE SRW System is adequate to perform the containment heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW containment heat removal function. Required Action A.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.6.6 should be entered for an inoperable containment cooling train. The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, the Completion Time associated with an inoperable containment cooling unit (3.6.6), and the low probability of a DBA occurring during this time period.

B.1

With one SRW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SRW System is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SRW System could result in loss of SRW function. Required Action B.1 is modified by a Note. The Note indicates that the applicable Conditions of LCO 3.8.1, should be entered if the inoperable SRW subsystem results in an inoperable diesel generator. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, and the low probability of a DBA occurring during this time period.

C.1 and C.2

If the SRW subsystem 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 3.7.6.1

Verifying the correct alignment for manual, power-operated, and automatic valves in the SRW flow path ensures that the proper flow paths exist for SRW 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 SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SRW components or systems may render those components inoperable but does not affect the OPERABILITY of the SRW System.

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

SR 3.7.6.2

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

SR 3.7.6.3

The SR verifies proper automatic operation of the SRW System pumps on an actual or simulated actuation signal (SIAS or CSAS). The SRW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR

B 3.7 PLANT SYSTEMS

B 3.7.7 Saltwater (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation or a normal shutdown, the SW System also provides this function for various safety-related and non-safety-related components. The safety-related function is covered by this LCO.

The SW System consists of two subsystems. Each subsystem contains one pump. A third pump, which is an installed spare, can be aligned to either subsystem. The safety-related function of each subsystem is to provide SW to two SRW heat exchangers, a CC heat exchanger, and an Emergency Core Cooling System (ECCS) pump room air cooler in order to transfer heat from these systems to the bay. Seal water for the non-safety-related circulating water pumps is supplied by both or either subsystems. The SW pumps provide the driving head to move SW from the intake structure, through the system and back to the circulating water discharge conduits. The system is designed such that each pump has sufficient head and capacity to provide cooling water such that 100% of the required heat load can be removed by either subsystem.

During normal operation, both subsystems in each unit are in operation with one pump running on each header and a third pump in standby. If needed, the standby pumps can be lined-up to either supply header. The SW flow through the SRW and CC heat exchangers is throttled to provide sufficient cooling to the heat exchangers, while maintaining total subsystem flow below a maximum value.

Additional information about the design and operation of the SW System, along with a list of the components served, is presented in Reference 1. During an accident, the SW System is required to remove the heat load from the SRW and ECCS pump room, and from the CC following an RAS.

APPLICABLE SAFETY ANALYSES

The most limiting event for the SW System is a LOCA. Operation of the SW System following a LOCA is separated into two phases, before the RAS and after the RAS. One subsystem can satisfy cooling requirements of both phases. After a LOCA but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler (as required). There is no required flow to the CC heat exchangers. When an RAS occurs, flow is throttled to the CC heat exchanger. Flow to each SRW heat exchanger is reduced while the system remains capable of providing the required flow to the ECCS pump room air coolers.

The SW System satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two SW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post-accident heat loads, assuming the worst single active failure occurs coincident with the loss of offsite power. Additionally, this system will also operate assuming the worst case passive failure post-RAS.

An SW subsystem is considered OPERABLE when:

a. The associated pump is OPERABLE; and

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

APPLICABILITY

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

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

ACTIONS

A.1

With one SW subsystem inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW subsystem is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the SW subsystem could result in loss of SW System function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions of LCO 3.8.1 should be entered if the inoperable SW subsystem results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6 should be entered if an inoperable SW subsystem results in an inoperable SDC. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SW subsystems 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

Verifying the correct alignment for manual, power-operated, and automatic valves in the SW System flow path ensures that the proper flow paths exist for SW 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 locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This surveillance test does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR is modified by a Note indicating that the isolation of the SW System components or systems may render those components inoperable but does not affect the OPERABILITY of the SW System.

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 SW System valves on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing. This surveillance test 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 test under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance test were performed with the reactor at power. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. Note: There are currently no SW valves with an Engineered Safety Feature Actuation System signal since automatic system reconfiguration during a LOCA is not required.

SR 3.7.7.3

The SR verifies proper automatic operation of the SW System pumps on an actual or simulated actuation signal (SIAS). The SW System is a normally operating system that cannot be fully actuated as part of the normal testing during normal operation. Operating experience has shown that these components usually pass the surveillance test when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.5.2.3, "Saltwater System"

B 3.7 PLANT SYSTEMS

B 3.7.8 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The CREVS is a shared system providing protection for both Unit 1 and Unit 2.

The CREVS consists of two trains, including redundant outside air intake ducts and redundant emergency recirculation filter trains that recirculate and filter the Control Room envelope (CRE) air and a CRE boundary that limits the inleakage of unfiltered air. The CREVS also has shared equipment, including an exhaust-to-atmosphere duct containing redundant isolation valves and a normally closed roof-mounted hatch, an exhaust-to-atmosphere duct from the kitchen and toilet area of the Control Room containing a single isolation valve, and common supply and return ducts in both the standby and emergency recirculation portions of the system. The shared equipment is considered to be a part of each CREVS train. Each CREVS emergency recirculation filter train consists of a prefilter, two high efficiency particulate air (HEPA) filters for removal of aerosols, an activated charcoal adsorber section for removal of elemental and organic iodine and a fan. Ductwork, valves or dampers, doors, and barriers also form part of the system. Instrumentation which actuates the system is addressed in LCOs 3.3.4 and 3.3.8.

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 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 DBA consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREVS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Actuation of the CREVS ensures the system is in the emergency recirculation mode of operation, ensures the unfiltered outside air intake and unfiltered exhaust-to-atmosphere valves are closed, and aligns the system for emergency recirculation of CRE air through the redundant trains of HEPA and charcoal filters. The prefilters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers. A control room recirculation signal (CRRS) initiates this filtered ventilation of the air supply to the CRE.

The air recirculating through the CRE is continuously monitored by a radiation detector. Detector output above the setpoint will cause actuation of the CREVS. The CREVS operation in maintaining the Control Room habitable is discussed in Reference 1, Section 9.8.2.3.

The redundant emergency recirculation filter train provides the required filtration should an excessive pressure drop develop across the other filter train. A normally closed hatch and double isolation valves are arranged in series to prevent a breach of isolation from the outside atmosphere, except for the exhaust from the Control Room kitchen and toilet areas. The CREVS is designed in accordance with Seismic Category I requirements.

The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a DBA without exceeding a 5 rem TEDE for the duration of the accident.

APPLICABLE SAFETY ANALYSES

The CREVS components are generally arranged in redundant safety-related ventilation trains although some equipment is shared between trains.

The CREVS provides automatic airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis fission product release presented in Reference 1, Section 14.24.

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release. The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the Control Room or from the remote shutdown panels.

The CREVS also provides automatically actuated airborne radiological protection for the Control Room operations, for the design basis fuel handling accident presented in Reference 1, Section 14.18, the control element assembly ejection event (Reference 1, Section 14.13, the main steam line break (Reference 1, Section 14.14), the steam generator tube rupture (Reference 1, Section 14.15), and the seized rotor event (Reference 1, Section 14.16). The fuel handling accident does not assume a single failure to occur.

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function (except for one valve in the shared duct between the Control Room and the emergency recirculation filter trains).

The CREVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

The CREVS is required to be OPERABLE to ensure that the Control Room is isolated and at least one emergency recirculation filter train is available, assuming a single active failure. Total system failure could result in exceeding a dose of 5 rem TEDE in the event of a large radioactive release.

The CREVS is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. For MODEs 1, 2, 3, and 4, redundancy is required and CREVS is considered OPERABLE when:

a. Both supply fans are OPERABLE;

b. Both recirculation fans are OPERABLE;

c. Both fans included in the emergency recirculation filter trains are OPERABLE;

d. Both HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions;

e. Ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained; and

f. The Control Room outside air intake can be isolated for the emergency recirculation mode of operation, assuming a single failure.

In order for the CREVS 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 analysis for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note which indicates that only one CREVS redundant component is required to be OPERABLE during movement of irradiated fuel assemblies, when both units are in MODEs 5 or 6, or defueled. Therefore, with both units in other than MODEs 1, 2, 3, or 4, redundancy is not required for movement of irradiated fuel assemblies and CREVS is considered OPERABLE when:

a. One supply fan is OPERABLE;

b. One recirculation fan is OPERABLE;

c. One fan included in the OPERABLE emergency recirculation filter train is OPERABLE;

d. One train of two HEPA filters and one charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

e. Associated ductwork, valves, and dampers are OPERABLE, such that air circulation can be maintained and the Control Room can be isolated for the emergency recirculation mode.

When implementing the Note (since redundancy is not required), only one of the two isolation valves in each outside air intake duct is required, and only one of the two isolation valves in the exhaust to atmosphere duct is required. However, the non-operating flow path must be capable of providing isolation of the Control Room from the outside atmosphere.

The LCO is modified by a second Note which indicates that only one CREVS train is required to be OPERABLE for the movement of irradiated fuel assemblies. Therefore, redundancy is not required for movement of irradiated fuel assemblies and only one CREVS train is required to be OPERABLE.

The LCO is modified by a third 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 condition, 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 proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when the need for CRE isolation is indicated.

APPLICABILITY

In MODEs 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

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

ACTIONS

A.1

With one or more ducts with one Control Room outside air intake isolation valve inoperable in MODEs 1, 2, 3, or 4, the OPERABLE Control Room outside air intake valve in each affected duct must be closed immediately. This places the

OPERABLE Control Room outside air intake isolation valve in each affected duct in its safety function required position.

B.1

With the toilet area exhaust isolation valve inoperable, action must be taken to restore OPERABLE status within 24 hours. In this Condition, the toilet area exhaust cannot be isolated, therefore, the valve must be restored to OPERABLE status. The 24 hour period allows enough time to repair the valve while limiting the time the toilet area is open to the atmosphere. The 24 hour Completion Time is based on the low probability of a DBA occurring during this time period.

C.1

With one exhaust to atmosphere isolation valve inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE exhaust to atmosphere isolation valve is adequate to isolate the Control Room. However, the overall reliability is reduced because a single failure in the OPERABLE exhaust to atmosphere isolation valve could result in loss of exhaust to atmosphere isolation valve function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining exhaust to atmosphere isolation valve to provide the required isolation capability.

D.1, D.2, and D.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 DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or 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 mitigation actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Required Action D.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. Compensatory measures are discussed in Reference 3. These compensatory measures may also be used as mitigating actions as required by Required Action D.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY. Actions must be taken within 24 hours to verify that, in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analysis of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional.

The 24 hour Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of the 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 DBA. 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.

E.1

With one CREVS train inoperable for reasons other than Conditions A, B, C, or D in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within seven days. In this Condition, the remaining OPERABLE CREVS subsystem is adequate to perform CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The seven day Completion Time is based on the low probability of a DBA occurring during this time period, and the ability of the remaining train to provide the required capability.

F.1 and F.2

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

G.1

Action G provides the actions to be taken when the Required Action and associated Completion Time of Condition B cannot be met or with one or more CREVS trains inoperable due to an inoperable CRE boundary. It requires the immediate suspension of movement of irradiated fuel assemblies. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies to a safe position. Since only one CREVS train must be OPERABLE for movement of irradiated fuel assemblies, the Required Action is applicable only to the required CREVS train.

H.1

If both CREVS trains are inoperable for reasons other than Conditions A, B, C, or D, or if one or more ducts have two outside air intake isolation valves inoperable, or if two exhaust to atmosphere isolation valves are inoperable, in MODEs 1, 2, 3, or 4, or during movement of irradiated fuel assemblies, the CREVS 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 and movement of irradiated fuel must be suspended immediately. This does not preclude the movement of fuel assemblies to a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each required CREVS filter train once every month provides an adequate check on this system.

The 31 day Frequency is based on the known reliability of the equipment, and the two filter train redundancy available.

SR 3.7.8.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.8.3

This SR verifies each CREVS train starts and operates on an actual or simulated actuation signal (CRRS). This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the CREVS is capable of starting and operating on an actual or simulated CRRS.

SR 3.7.8.4

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 the CRE occupants calculated in the licensing basis analysis of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analysis of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition E must be entered. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequences 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.

REFERENCES

1. UFSAR

2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978

3. Regulatory Guide 1.196, Revision 0, "Control Room Habitability at Light-Water Nuclear Power Reactors," May 2003

B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Temperature System (CRETS)

BASES

BACKGROUND	The CRETS provides temperature control for the Control Room following isolation of the Control Room. The CRETS is a shared system which is supported by the CREVS, since the CREVS must be operating in the emergency recirculation mode for CRETS to perform its safety function. The CRETS consists of two independent, redundant trains that provide cooling of recirculated Control Room air. Each train consists of cooling coils, instrumentation, and controls to provide for Control Room temperature control. The CRETS is a subsystem providing air temperature control for the Control Room. The CRETS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. A single train will provide the required temperature control to maintain the Control Room below 104°F. The CRETS operation to maintain the Control Room temperature is discussed in Reference 1.
APPLICABLE SAFETY ANALYSES	The design basis of the CRETS is to maintain temperature of the Control Room environment throughout 30 days of continuous occupancy. The CRETS components are arranged in redundant safety-related trains. During emergency operation, the CRETS maintains the temperature below 104°F. A single active failure of a component of the CRETS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for Control Room temperature control. The CRETS is designed in accordance with Seismic Category I requirements. The CRETS is capable of removing sensible and latent heat loads from the Control Room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY. The CRETS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two independent and redundant trains of the CRETS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train following isolation of the Control Room. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident requiring isolation of the Control Room.

The CRETS is considered OPERABLE when the individual components that are necessary to maintain the Control Room temperature are OPERABLE. The required components include the cooling coils and associated temperature control instrumentation. In addition, the CRETS must be OPERABLE to the extent that air circulation can be maintained.

For MODEs 1, 2, 3, and 4, redundancy is required and both trains must be OPERABLE. The LCO is modified by a Note which indicates that only one CRETS train is required to be OPERABLE for the movement of irradiated fuel assemblies. Therefore, redundancy is not required for movement of irradiated fuel assemblies and only one CRETS train is required to be OPERABLE.

APPLICABILITY

In MODEs 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CRETS must be OPERABLE to ensure that the Control Room temperature will not exceed equipment OPERABILITY requirements following isolation of the Control Room.

ACTIONS

A.1

With one CRETS train inoperable in MODEs 1, 2, 3, or 4, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRETS train is adequate to maintain the Control Room temperature within limits. The 30 day Completion Time is reasonable, based on the low probability of an event occurring requiring Control Room isolation, consideration that the remaining train can provide the required capabilities, and the alternate safety or non-safety-related cooling means that are available.

B.1 and B.2

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

C.1

If both CRETS trains are inoperable in MODEs 1, 2, 3, or 4, or during movement of irradiated fuel assemblies, the CRETS may not be capable of performing the intended function and the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately and movement of irradiated fuel must be suspended immediately. This does not preclude the movement of fuel assemblies to a safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies each required CRETS train has the capability to maintain Control Room temperature $\leq$ 104°F for $\geq$ 12 hours in the recirculation mode. During this test, the backup Control Room air conditioner is to be de-energized. This SR consists of a combination of testing. A 24 month Frequency is appropriate, since significant degradation of the CRETS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 9.8.2.3, "Auxiliary Building Ventilating Systems"

B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool Exhaust Ventilation System (SFPEVS)

BASES

BACKGROUND

The SFPEVS exhausts airborne radioactive particulates and gases from the area of the fuel pool into the plant ventilation stack following a fuel handling accident involving recently irradiated fuel.

The SFPEVS consists of two independent, redundant exhaust fans. Ductwork, valves or dampers, and instrumentation also form part of the system. The SFPEVS is supplied power by one non-safety-related power supply.

The SFPEVS is operated during normal unit operations. When movement of the air is required (i.e., during movement of recently irradiated fuel assemblies in the Auxiliary Building), normal air discharges from the fuel handling area in the Auxiliary Building.

The SFPEVS is discussed in Reference 1, Sections 9.8.2.3 and 14.18, because it may be used for normal, as well as post-accident ventilation.

APPLICABLE SAFETY ANALYSES

The SFPEVS is designed to mitigate the consequences of a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 55 days), in which all rods in the fuel assembly are assumed to be damaged. The analysis of the fuel handling accident is given in Reference 1, Section 14.18. The DBA analysis of the fuel handling accident assumes that the SFPEVS is functional and exhausts airborne radioactive particulates and gases from the fuel pool area into the plant ventilation stack. The analysis follows the guidance provided in Reference 2.

The SFPEVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

Two exhaust fans and other equipment listed in the Background Section are required to be OPERABLE and in operation.

The SFPEVS is considered OPERABLE when the individual components necessary to direct exhaust into the ventilation stack are OPERABLE. The SFPEVS is considered OPERABLE when its associated:

a. Fans are OPERABLE; and

b. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

The SFPEVS is considered in operation when an OPERABLE exhaust fan is in operation.

APPLICABILITY

During movement of recently irradiated fuel assemblies in the Auxiliary Building, the SFPEVS is required to be OPERABLE and in operation to mitigate the consequences of a fuel handling accident involving handling recently irradiated fuel by minimizing the atmospheric dispersion to the Control Room. Due to radioactive decay, the SFPEVS is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 55 days).

ACTIONS

A.1 and A.2

When one SFPEVS exhaust fan is inoperable, action must be taken to verify an OPERABLE SFPEVS train is in operation, or movement of recently irradiated fuel assemblies in the Auxiliary Building must be suspended. One OPERABLE SFPEVS train consists of one OPERABLE exhaust fan. This ensures the proper equipment is operating for the Applicable Safety Analysis.

B.1

When there is no OPERABLE SFPEVS train or there is no OPERABLE SFPEVS train in operation during movement of recently irradiated fuel assemblies in the Auxiliary Building, action must be taken to place the unit in a condition in which the LCO does not apply. This Action involves immediately suspending movement of recently irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

The SR requires verification every 12 hours that the SFPEVS is in operation. Verification includes verifying that one exhaust fan is operating and discharging into the ventilation stack. The Frequency of 12 hours is sufficient considering that the operators will be focused on the movement of recently irradiated fuel assemblies within the Auxiliary Building. Thus, if anything were to occur to cause cessation of operation of the SFPEVS, it would be quickly identified.

SR 3.7.11.2

Deleted.

SR 3.7.11.3

This SR verifies the integrity of the spent fuel storage pool area. The ability of the spent fuel storage pool area to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the SFPEVS. During operation, the spent fuel storage pool area is designed to maintain a slight negative pressure in the spent fuel storage pool area, with respect to adjacent areas, to ensure that exhausted air is directed to the ventilation stack.

This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the SFPEVS is capable of maintaining a negative pressure.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

B 3.7 PLANT SYSTEMS

B 3.7.12 Penetration Room Exhaust Ventilation System (PREVS)

BASES

BACKGROUND

The PREVS filters air from the penetration room.

The PREVS consists of two independent and redundant trains. Each train consists of a prefilter, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation following receipt of a containment isolation actuation signal.

The PREVS is a standby system, which may also operate during normal unit operations. During emergency operations, the PREVS dampers are realigned, and fans are started to initiate filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the penetration room, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The PREVS is discussed in Reference 1, Section 6.6.2, as it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The design basis of the PREVS is established by the Maximum Hypothetical Accident. The system is credited with filtering the radioactive material released through the containment vent when the line is open. Also commensurate with the guidance in Reference 3, a conservative bypass fraction from the Containment to the penetration rooms is assumed. Following a LOCA, the containment isolation signal will start both of the fans associated with the PREVS, filtering the exhaust through the HEPA and charcoal filters, and directing the exhaust into the ventilation stack. The analysis of the effects and consequences of a Maximum Hypothetical Accident are presented in Reference 1, Section 14.24 and follows the guidance presented in Reference 4.

As a layer of defense, the Penetration Room Exhaust Ventilation System also provides filtered ventilation of radioactive materials leaking from ECCS equipment within the penetration room following an accident, however, credit for this feature was not assumed in the accident analysis (Reference 1, Section 14.24).

The PREVS satisfies 10 CFR 50.36(c)(2)(ii), Criterion 3.

LCO

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

The PREVS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREVS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. High efficiency particulate air filter and charcoal adsorber are not excessively restricting flow, and are capable of performing the filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and circulation can be maintained.

APPLICABILITY

In MODEs 1, 2, and 3, the PREVS is required to be OPERABLE to mitigate the potential radioactive material release from a Maximum Hypothetical Accident.

In MODEs 4, 5, and 6, the PREVS is not required to be OPERABLE, since the RCS temperature and pressure are low and there is insufficient energy to result in the conditions assumed in the accident analysis.

ACTIONS

A.1

With one PREVS train inoperable, action must be taken to restore OPERABLE status within seven days. During this time period, the remaining OPERABLE train is adequate to perform the PREVS function. The seven day Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the consideration that the remaining train can provide the required capability.

B.1 and B.2

If the inoperable 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 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 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

The test is performed by initiating the system from the Control Room, ensuring flow through the HEPA filter and charcoal adsorber train, and verifying this system operates for ≥ 15 minutes. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies the performance of PREVS filter testing in accordance with the VFTP. The PREVS filter tests are in accordance with portions of Reference 2. The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each PREVS train starts and operates on an actual or simulated actuation signal (Containment Isolation Signal). This test is conducted on a 24 month Frequency. This Frequency is adequate to ensure the PREVS is capable of starting and operating on an actual or simulated Containment Isolation Signal.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," March 1978
3. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003
4. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the SFP 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 SFP design is given in Reference 1, Section 9.7.2, and the SFP Cooling and Cleanup System is given in Reference 1, Section 9.4.1. The assumptions of the fuel handling accident are given in Reference 1, Section 14.18.

APPLICABLE SAFETY ANALYSES

Per Reference 2, the Fuel Handling Accident (FHA) analysis may assume a total iodine decontamination factor of 200 based on a minimum water depth of 23 feet. The minimum water level requirement ensures that sufficient water depth is available to remove 99.5% of gap activity, which is comprised of 16% I-131 and 10% of all other iodine isotopes released from the rupture of an irradiated fuel assembly.

The Technical Specifications requirement of 21.5 feet of water above fuel assemblies seated in the SFP storage racks is sufficient to preserve the required 23 feet of water because an FHA was assumed to occur as a fuel assembly strikes the bottom of the SFP.

When assemblies are placed on rack spacers with their upper end fittings removed, an FHA caused by a dropped heavy object would result in a lower decontamination factor based on reduced water coverage. A revised decontamination factor of 120 for an FHA during reconstitution or inspection with 20.4 feet of water between the top of the pin and the surface of the water was computed for an assembly placed on a 20.5 inch rack spacer with its upper end fitting removed. Note that this is very conservative, since normal level control will result in at least 21.5 feet of water above exposed fuel pins. This results in a 99.17% removal rate.

The SFP water level satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Reference 1, Section 14.18). As such, it is the minimum required for fuel storage, reconstitution, and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the SFP 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 an accident cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude moving a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODEs 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, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the SFP must be checked periodically. The seven day Frequency is appropriate, because the volume in the pool is normally stable. Water level changes are controlled by unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the SFP is normally at equilibrium with that of the refueling canal.

REFERENCES

1. UFSAR
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND	Activity in the secondary coolant results from steam generator tube outleakage from the RCS. Under steady state conditions, the activity is primarily iodines with relatively short half lives, and thus is an indication of current conditions. During transients, DOSE EQUIVALENT 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 is lower than the activity value that might be expected from a 100 gallons per day tube leak (LCO 3.4.13) of primary coolant at the limit of 0.5 μCi/gm (LCO 3.4.15). The main SLB is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and reactor coolant LEAKAGE via flashing directly to the environment through the main steam gooseneck.
APPLICABLE SAFETY ANALYSES	The accident analysis of the main SLB, as discussed in Reference 1, assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This secondary activity, together with the Technical Specification primary system activity, and failed fuel activity, is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis shows that the radiological consequences of a main SLB do not exceed the acceptance criteria given in References 1 and 2. With the loss of offsite power post-main SLB, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through MSSVs and ADVs. The AFW System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant

temperature and pressure have decreased sufficiently for the SDC System to complete the cooldown.

Other accidents or transients, such as a steam generator tube rupture, a seized rotor event, and a control element assembly ejection event, involve a partial release of the secondary activity via steam release to the atmosphere via the ADVs and MSSVs. These releases contribute to the offsite and Control Room doses listed in Reference 1, Section 14. These accident analyses show that the radiological consequences of a DBA do not exceed the acceptance criteria given in References 1 and 2.

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

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 limits the radiological consequences of a DBA to the acceptance criteria given in Reference 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 DBA.

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.

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

BASES

secondary specific activity cannot be restored to within limits in 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 3.7.14.1

This SR ensures that the secondary specific activity is within the limits of the accident analysis. A gamma isotope 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. UFSAR, Chapter 14, "Safety Analysis"
2. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

B 3.7 PLANT SYSTEMS

B 3.7.15 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The MFIVs isolate MFW flow to the secondary side of the steam generators following a HELB. The consequences of HELBs occurring in the main steam lines or in the MFW lines downstream of the MFIVs will be mitigated by their closure. Closure of the MFIVs effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for SLBs /or feedwater line breaks (FWLBs) inside the Containment Structure upstream of the reverse flow check valve, and reducing the cooldown effects for SLBs.

The MFIVs isolate the non-safety-related portions from the safety-related portion of the system. In the event of a secondary side pipe rupture inside the Containment Structure upstream of the reverse flow check valve, the valves limit the quantity of high energy fluid that enters the Containment Structure through the break.

One MFIV is located on each MFW line, outside, but close to, the Containment Structure. The MFIVs are located so that AFW may be supplied to a steam generator following MFIV closure. The piping volume from the valve to the steam generator must be accounted for in calculating mass and energy releases.

The MFIVs close on receipt of a steam generator isolation signal generated by low steam generator pressure. The steam generator isolation signal also actuates the MSIVs to close. The MFIVs may also be actuated manually. In addition, the MFIVs reverse flow check valve inside the Containment Structure is available to isolate the feedwater line penetrating the Containment Structure, and to ensure that the consequences of events do not exceed the capacity of the Containment Cooling System.

A description of the MFIVs operation on receipt of an steam generator isolation signal is found in Reference 1.

BASES

APPLICABLE SAFETY ANALYSES	The design basis of the MFIVs is established by the analysis for the large SLB. It is also influenced by the accident analysis for the large FWLB. Failure of an MFIV to close following an SLB or FWLB can result in additional mass and energy to the steam generator's contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. The MFIVs satisfy 10 CFR 50.36(c)(2)(ii), Criterion 3.
LCO	This LCO ensures that the MFIVs will isolate MFW flow to the steam generators. Following an FWLB or SLB, these valves will also isolate the non-safety-related portions from the safety-related portions of the system. This LCO requires that one MFIV in each feedwater line be OPERABLE. The MFIVs are considered OPERABLE when the isolation times are within limits, and are closed on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to the Containment Structure following an SLB or FWLB inside the Containment Structure. Failure to meet the LCO can also add additional mass and energy to the steam generators contributing to cooldown.
APPLICABILITY	The MFIVs must be OPERABLE whenever there is significant mass and energy in the RCS and steam generators. In MODEs 1, 2, and 3, the MFIVs are required to be OPERABLE in order to limit the amount of available fluid that could be added to the Containment Structure in the case of a secondary system pipe break inside the Containment Structure. In MODEs 4, 5, and 6, steam generator energy is low.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one MFIV inoperable, action must be taken to restore the valve to OPERABLE status within 72 hours.

The 72 hour Completion Time takes into account the isolation capability afforded by the MFW regulating valves, and tripping of the MFW pumps, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

B.1 and B.2

If the MFIVs cannot be restored to OPERABLE status in 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 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 3.7.15.1

This SR ensures the closure time for each MFIV is $\leq$ 65 seconds by manual isolation. The MFIV closure time is assumed in the accident and containment analyses.

The Frequency is in accordance with the Inservice Testing Program. The MFIVs are tested during each refueling outage in accordance with Reference 2, and sometimes during other cold shutdown periods. The Frequency demonstrates the valve closure time at least once per refueling cycle. Operating experience has shown that these components usually pass the surveillance test when performed.

BASES

REFERENCES	1.	UFSAR, Section 14.4.2, "Sequence of Events"
	2.	ASME Code for Operation and Maintenance of Nuclear Power Plants

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND	Fuel assemblies are stored in the spent fuel racks in accordance with criteria based on 10 CFR 50.68. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical) at a 95% probability, 95% confidence level, if flooded with unborated water. In addition, the maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to 5.0 weight percent.
APPLICABLE SAFETY ANALYSES	The criticality analyses were done such that the criteria of 10 CFR 50.68 are met. Boron dilution events are credible, postulated accidents, when credit for soluble boron is taken. The minimum SFP boron concentration in this Technical Specification supports the initial boron concentration assumption in the dilution calculations. For other non-dilution accident scenarios, the double contingency principle of ANSI N 16.1-1975 requires two unlikely, independent concurrent events to produce a criticality accident and thus allows credit for the nominal soluble boron concentration, as defined in LCO 3.7.16. The concentration of dissolved boron in the SFPs satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the SFPs.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the SFPs.

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. 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 MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the SFPs is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

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

REFERENCES

None

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND	**This Technical Specification applies to the Unit 2 SFP only.** The spent fuel storage facility was originally designed to store either new (non-irradiated) nuclear fuel assemblies or burned (irradiated) fuel assemblies in a vertical configuration underwater, assuming credit for Boraflex poison sheets but assuming no credit for soluble boron or burnup. The spent fuel storage cells are installed in parallel rows with center-to-center spacing of 10 3/32 inches and with Boraflex sheets between adjacent assemblies. This spacing was sufficient to maintain k_{eff} $\leq$ 0.95 for spent fuel of enrichments up to 4.52 wt% for standard fuel design and up to 4.30 wt% for Value Added Pellet (VAP) fuel design. The burnup and enrichment requirements of LCO 3.7.17(a) ensures that the multiplication factor (k_{eff}) for the rack in the SFP is less than the 10 CFR 50.68 regulatory limit with the VAP fuel design, ranging in enrichment from 2.0 to 5.0 wt%, with burnup credit, with partial credit for soluble boron, but without Boraflex credit. The soluble boron credit will be limited to 350 ppm including all biases and uncertainties. For fuel assemblies which do not satisfy the burnup and enrichment requirements of LCO 3.7.17(a), the fuel assemblies may be stored in the Unit 2 SFP if surrounded on all four adjacent faces by empty rack cells or other non-reactive materials per LCO 3.7.17(b).
APPLICABLE SAFETY ANALYSES	The Unit 2 spent fuel storage facility is designed to conform to the requirements of 10 CFR 50.68 by use of adequate spacing, soluble boron credit, and burnup credit. The SFP storage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the Unit 2 SFP are in accordance with Figure 3.7.17-1 and ensure that the Unit 2 SFP meets the requirements of 10 CFR 50.68. The restrictions are consistent with the criticality safety analysis performed for the Unit 2 SFP. Fuel assemblies not meting the criteria of Figure 3.7.17-1 may be

BASES

stored in the Unit 2 SFP in a checkboard pattern in accordance with LCO 3.7.17(b).

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the Unit 2 SFP.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in Unit 2 SFP is not in accordance with Figure 3.7.17-1 or LCO 3.7.17(b), immediate action must be taken to make the necessary fuel assembly movement(s) to bring the fuel assembly configuration into compliance with Figure 3.7.17-1 or LCO 3.7.17(b).

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.17-1 for LCO 3.7.17(a). This Surveillance Requirement does not address fuel assemblies stored in the Unit 2 SFP in accordance with LCO 3.7.17(b). This will ensure compliance with Specification 4.3.1.1.

REFERENCES

None