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) downstream of the MSIV. MSIV closure terminates flow from the unaffected (intact) steam generator for breaks upstream of the other MSIV.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the Main Steam Safety Valves (MSSVs), atmospheric dump valves, and auxiliary feedwater 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, turbine bypass valve, and other auxiliary steam supplies from the steam generators, assuming the normally closed MSIV bypass valves are closed. The MSIV bypass valves do not receive an isolation signal and might be open during zero power conditions.

The MSIVs close on isolation signals generated by either Steam Generator Low Pressure or Containment High Pressure. The MSIVs fail closed on loss of air. The isolation signal also actuates the Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves to close. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the Main Steam Line Break (MSLB) inside containment, as discussed in the FSAR, Section 14.18 (Ref. 2). It is also influenced by the accident analysis of the MSLB events presented in the FSAR, Section 14.14 (Ref. 3). The MSIVs are swing disc check valves. The inherent characteristic of this type of valve allows for reverse flow through the valve on a differential pressure even if the valve is closed. In the event of an MSLB, if the MSIV associated with the unaffected steam generator fails to close, both steam generators may blowdown. This failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a double steam generator blowdown event, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

There are three different limiting MSLB cases that have been evaluated, one for fuel integrity and two for containment analysis (one for containment temperature and one for containment pressure). The limiting case for containment temperature is the hot full power MSLB inside containment following a turbine trip. At hot full power, the stored energy in the primary coolant is maximized.

The limiting case for the containment analysis for containment pressure and fuel integrity is the hot zero power MSLB inside containment. At zero power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Reverse flow due to the open MSIV bypass valves, contributes to the total release of the additional mass and energy. With the most reactive control rod assumed stuck in the fully withdrawn position, there is an increased possibility that the core will return to power. The core is ultimately shut down by a combination of doppler feedback, steam generator dryout, and borated water injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different MSLB events against different acceptance criteria. The MSLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The MSLB inside containment 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 a turbine trip.

With offsite power available, the primary coolant pumps continue to circulate coolant through the steam generators, maximizing the Primary Coolant System (PCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Safety Injection (HPSI) pumps, is delayed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

a. An MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator.

b. A break outside of containment 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 PCS 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 turbine bypass valve will also terminate on closure of the MSIVs.

d. A steam generator tube rupture. For this scenario, closure of the MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

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

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 signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100.11 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when both MSIVs are closed and deactivated when there is significant mass and energy in the PCS and steam generators. When the MSIVs are closed, they are already performing their safety function. Deactivation can be accomplished by the removal of the motive force (e.g., air) to the valve to prevent valve opening.

BASES

APPLICABILITY (continued)

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

A.1

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 plant hot. The 8 hour Completion Time is reasonable, considering the probability of an accident occurring during the time period that would require closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment.

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging plant 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 8 hour Completion Time is consistent with that allowed in Condition A.

BASES

ACTIONS

C.1 and C.2 (continued)

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 once per 7 days 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. As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.7.2 C.2 must be initially performed within 7 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status, or closed, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is ≤ 5.0 seconds on an actual or simulated actuation signal from each train under no flow conditions. Specific signals (e.g., Containment High Pressure, Steam Generator Low Pressure, handswitch) are tested under Section 3.3, "Instrumentation." The MSIV closure time is assumed in the MSLB and containment analyses. This SR is normally performed upon returning the plant to operation following a refueling outage. The MSIVs are not tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODES 1 and 2.

The Frequency for this SR is every 18 months. This 18 month 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 at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.2
2. FSAR, Section 14.18
3. FSAR, Section 14.14
4. 10 CFR 100.11
5. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs and MFRV bypass valves in conjunction with feed pump speed, control Main Feedwater (MFW) flow to the steam generators for level control during normal plant operation. The valves also isolate MFW flow to the secondary side of the steam generators following a High Energy Line Break (HELB). Closure of the MFRVs and MFRV bypass valves terminates flow to both steam generators. Closure of the MFRV and MFRV bypass valve effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for Main Steam Line Breaks (MSLBs) inside containment, and reducing the cooldown effects.

The MFRVs and MFRV bypass valves isolate MFW in the event of a secondary side pipe rupture inside containment to limit the quantity of high energy fluid that enters containment through the break. Controlled addition of Auxiliary Feedwater (AFW) is provided by a separate piping system.

One MFRV and one MFRV bypass valve are located on each MFW line outside containment. The piping volume from the valves to the steam generator must be accounted for in calculating mass and energy releases following an MSLB.

The MFRVs and MFRV bypass valves close on receipt of an isolation signal generated by either; steam generator low pressure from its respective steam generator, or containment high pressure. These isolation signals also actuate the Main Steam Isolation Valves (MSIVs) to close. The MFRVs and MFRV bypass valves may also be actuated manually. The MFRVs and MFRV Bypass valves are non-safety grade valves located on non-safety grade piping that fail "as-is" on a loss of air. If required, MFW isolation can be accomplished using manually operated valves upstream or downstream of the MFRVs and MFRV Bypass valves. In addition, each MRFV is equipped with a handwheel that can be used to isolate this MFW flowpath.

A description of the MFRVs and MFRV bypass valves is found in the FSAR, Section 10.2.3 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

Closure of the MFRVs is an assumption in the MSLB containment response analysis. Closure of the MFRVs and MFRV bypass valves is also assumed in the MSLB core response (DNB) analysis.

Failure of an MFRV or MFRV bypass valve to close following an MSLB can result in additional mass and energy to the steam generators contributing to cooldown. This failure also results in additional mass and energy releases following an MSLB event. However, this failure was not analyzed as part of the original licensing basis of the plant. As such, a Probabilistic Risk Assessment and cost benefit analysis were performed to determine if a facility modification was needed. The results of the analysis as described in an NRC Safety Evaluation dated February 28, 1986 concluded that a single steam generator blowdown event with continued feedwater, although more severe than the MSLB used in the original licensing basis of the plant, is not expected to result in unacceptable consequences. Furthermore, the NRC evaluation demonstrated that the potential offsite dose consequences are low and that modifications would not provide a cost beneficial improvement to plant safety.

The MFRVs and MFRV bypass valves satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO ensures that the MFRVs and MFRV bypass valves will isolate MFW flow to the steam generators following an MSLB. This LCO requires that both MFRVs and both MFRV bypass valves be OPERABLE. The MFRVs and MFRV bypass valves are considered OPERABLE when the isolation times are within limits, and are closed on an isolation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an MSLB inside containment.

BASES

APPLICABILITY

All MFRVs and MFRV bypass valves must be OPERABLE, or either closed and deactivated, or isolated by closed manually actuated valves, whenever there is significant mass and energy in the Primary Coolant System and steam generators.

In MODES 1, 2, and 3, the MFRVs or MFRV bypass valves are required to be OPERABLE, except when both MFRVs and both MFRV bypass valves are either closed and deactivated, or isolated by closed manually actuated valves, in order to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are either closed and deactivated, or isolated by closed manually actuated valves, they are already performing their safety function.

Once the valves are closed, deactivation can be accomplished by the removal of the motive force (e.g., electrical power, air) to the valve to prevent valve opening. Closing another manual valve in the flow path either remotely (i.e., control room hand switch) or locally by manual operation satisfies isolation requirements.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFRVs and MFRV bypass valves are not required to be OPERABLE.

ACTIONS

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

A.1 and A.2

With one or more MFRV or MFRV bypass valve inoperable, action must be taken to close or isolate the inoperable valve(s) within 8 hours. When these valve(s) are closed or isolated, they are performing their required safety function (e.g., to isolate the line).

The 8 hour Completion Time is reasonable to close the MFRV or MFRV bypass valve, which includes performing a controlled plant shutdown to a condition that supports isolation of the affected valve(s). As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as “once per . . .” however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance.

BASES

ACTIONS (continued)

A.1 and A.2 (continued)

Therefore, while Required Action 3.7.3 A.2 must be initially performed within 7 days without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

B.1 and B.2

If the MFRVs or MFRV bypass valves cannot be restored to OPERABLE status, closed, or isolated in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies the closure time for each MFRV and MFRV bypass valve is ≤ 22.0 seconds on an actual or simulated actuation signal. Specific signals (e.g., steam generator low pressure and containment high pressure) are tested under Section 3.3, "Instrumentation." The MFRV and MFRV bypass valves closure times are bounding values assumed in the MSLB containment response and core response (DNB) analyses (Refs. 3 and 4). This SR is normally performed upon returning the plant to operation following a refueling outage. The MFRVs and MFRV bypass valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not stroke tested at power, they are exempt from the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.

The Frequency is 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency.

BASES

REFERENCES

1. FSAR, Section 10.2.3
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400
3. FSAR, Section 14.18.2
4. FSAR, Section 14.14

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the plant to Shutdown Cooling (SDC) System entry conditions, should the preferred heat sink via the turbine bypass valve to the condenser not be available, as discussed in the FSAR, Section 10.2 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the Condensate Storage Tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the turbine bypass valve.

Four ADVs are provided, two per steam generator. One ADV per steam generator is required following an event rendering one steam generator unavailable for Primary Coolant System (PCS) heat removal.

The ADVs are provided with upstream manual isolation valves to provide a means of isolation in the event an ADV spuriously opens, or fails to close during use. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized gas supply from the Bulk Nitrogen System that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ADVs. The nitrogen backup is not required for ADV OPERABILITY. A description of the nitrogen backup is found in the FSAR, Section 9.5.2 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is to prevent lifting of the Main Steam Safety Valves (MSSVs) following a turbine and reactor trip, and to provide the capability to cool the plant to SDC System entry conditions when condenser vacuum is lost. A cooldown rate of approximately 75°F per hour is obtainable by one or both steam generators. This design is adequate to cool the plant to SDC System entry conditions with only one ADV and one steam generator, utilizing the cooling water supply available in the Condensate Storage and Supply system.

BASES

APPLICABLE SAFETY ANALYSES (continued)	In certain accident analyses presented in the FSAR, the ADVs are assumed to be used by the operator to cool down the plant to SDC System entry conditions for accidents accompanied by a loss of offsite power. The ADVs are credited for cooldown during a Steam Generator Tube Rupture (SGTR) event. Prior to the operator action, the Main Steam Safety Valves (MSSVs) are used to maintain steam generator pressure and temperature at or below the MSSV setpoint for 30 minutes following the initiation of an event. The ADVs are also credited in selected safety analyses when the Auxiliary Feedwater (AFW) System is required to operate. If AFW pump P-8C is used, operator action may be required to either trip the four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required AFW flowrate to the steam generators assumed by the loss of feedwater analysis. The ADVs are equipped with manual isolation valves in the event an ADV spuriously opens, or fails to close during use. The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	One ADV is required to be OPERABLE on each steam generator to ensure that at least one ADV is OPERABLE to conduct a plant cooldown following an event in which one steam generator becomes unavailable. A closed manual isolation valve does not render its ADV inoperable, since operator action time to open the manual isolation valve is supported in the accident analysis. Failure to meet the LCO can result in the inability to cool the plant to SDC System entry conditions following an event in which the condenser is unavailable for use with the turbine bypass valve. An ADV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand from either the control room or Hot Shutdown Panel (C-33).
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE. In MODES 5 and 6, there are no credible transients requiring ADVs.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one required ADV inoperable, action must be taken to restore the ADV to OPERABLE status within 7 days. The 7 day Completion Time takes into account the redundant capability afforded by the remaining OPERABLE ADV, and a nonsafety grade backup in the turbine bypass valve and MSSVs.

B.1

With two required ADVs inoperable, action must be taken to restore one of the ADVs to OPERABLE status. As the manual isolation valve can be closed to isolate an ADV, some repairs may be possible with the plant at power. The 24 hour Completion Time is reasonable to repair inoperable ADVs, based on the availability of the turbine bypass valve and MSSVs, and the low probability of an event occurring during this period that requires the ADVs.

C.1 and C.2

If the ADVs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon the steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the PCS, the ADVs must be able to be cycled through their full range. This SR ensures the ADVs are tested through a full control cycle at least once per 18 months. Performance of inservice testing or use of an ADV during a plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.2

2. FSAR, Section 9.5.2

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Primary Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction through a common suction line from the Condensate Storage Tank (CST) (LCO 3.7.6, "Condensate Storage and Supply") and pump to the steam generator secondary side via two separate and independent flow paths to a common AFW supply header for each steam generator. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the Main Steam Safety Valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") or Atmospheric Dump Valves (ADVs) (LCO 3.7.4, "Atmospheric Dump Valves (ADVs)"). If the main condenser is available, steam may be released via the turbine bypass valve.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into two trains. One train (A/B) consists of a motor driven pump (P-8A) and the turbine driven pump (P-8B) in parallel, the discharges join together to form a common discharge. The A/B train common discharge separates to form two flow paths, which feed each steam generator via each steam generator's AFW penetration. The second motor driven pump (P-8C) feeds both steam generators through separate flow paths via each steam generator AFW penetration and forms the other train (C). The two trains join together at each AFW penetration to form a common supply to the steam generators. Each AFW pump is capable of providing 100% of the required capacity to the steam generators as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

Each motor driven AFW pump is powered from an independent Class 1E power supply, and feeds both steam generators.

BASES

BACKGROUND (continued)

The steam turbine driven AFW pump receives steam from the steam generator E-50A main steam header upstream of the Main Steam Isolation Valve (MSIV). The steam supply valve receives an open signal from the Auxiliary Feedwater Actuation Signal (AFAS) instrumentation. The turbine driven AFW pump feeds both steam generators through the same flow paths as motor driven AFW pump P-8A.

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

The AFW System supplies feedwater to the steam generators during normal plant startup, shutdown, and hot standby conditions.

The AFW System is designed to supply sufficient water to the steam generators to remove decay heat with steam generator pressure at the setpoint of the MSSVs, with exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip the four Primary Coolant Pumps (PCPs), start an additional AFW pump, or reduce steam generator pressure. This will allow the required flowrates to the steam generators that are assumed in the safety analyses. Subsequently, the AFW System supplies sufficient water to cool the plant to SDC entry conditions, and steam is released through the ADVs, or the turbine bypass valve if the condenser is available.

The AFW System actuates automatically on low steam generator level by an AFAS as described in LCO 3.3.3, "Engineered Safety Feature (ESF) Instrumentation" and 3.3.4, "ESF Logic." The AFAS initiates signals for starting the AFW pumps and repositioning the valves to initiate AFW flow to the steam generators. The actual pump starts are on an "as required" basis. P-8A is started initially, if the pump fails to start, or if the required flow is not established in a specified period of time, P-8C is started. If P-8A and P-8C do not start, or if required flow is not established in a specified period of time, then P-8B is started.

The AFW System is discussed in the FSAR, Section 9.7 (Ref. 1).

BASES

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% with the exception of AFW pump P-8C. If AFW pump P-8C is used, operator action may be required to either trip the four PCPs, start an additional AFW pump or reduce steam generator pressure. This will allow the required flowrate to the steam generators that are assumed in the safety analyses.

The limiting Design Basis Accident for the AFW System is a loss of normal feedwater.

In addition, the minimum available AFW flow and system characteristics impact the analysis of a small break loss of coolant accident.

The AFW System design is such that it can perform its function following loss of normal feedwater combined with a loss of offsite power with one AFW pump injecting AFW to one steam generator.

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

LCO

This LCO requires that two AFW trains be OPERABLE to ensure that the AFW System will perform the design safety function to mitigate the consequences of accidents that could result in overpressurization of the primary coolant pressure boundary. Three independent AFW pumps, in two diverse trains, ensure availability of residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two pumps from independent emergency buses. The third AFW pump is powered by a diverse means, a steam driven turbine supplied with steam from a source not isolated by the closure of the MSIVs.

BASES

LCO (continued)

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 two motor driven AFW pumps be OPERABLE in two diverse paths, each supplying AFW to both steam generators. Prior to making the reactor critical during a plant startup, the turbine driven AFW pump shall be OPERABLE and capable of supplying AFW flow to both steam generators. When steam generator pressure is reduced, it is not required to have design inlet pressure available to the turbine driver in order to declare the turbine driven AFW pump OPERABLE. As steam generator pressure drops, the required AFW pump discharge head decreases accordingly. The reduced steam generator pressure available at lower temperatures in MODE 3 does not inhibit the turbine driven AFW pump's ability to feed the steam generator (Ref. 3). The piping, valves, instrumentation, and controls in the required flow paths shall also be OPERABLE.

The LCO is modified by three Notes. Note one indicates that only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of reduced heat removal requirements, the short period of time in MODE 4 during which AFW is required, and the insufficient steam pressure available in MODE 4 to power the turbine driven AFW pump. Note two states that the turbine driven AFW pump is only required to be made OPERABLE prior to making the reactor critical. It is required to be OPERABLE during subsequent MODE 1, 2, and 3 operation. This allowance is needed to provide sufficient steam pressure to perform turbine and pump testing. Note three indicates that any two AFW pumps may be placed in manual mode for the purpose of testing, for not more than 4 hours. In this situation, the third AFW pump would still be available in the event of a plant transient. The two pumps that are in manual could be used at the discretion of the operator.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE and to function in the event that the main feedwater is lost. In addition, the AFW System is required to supply enough makeup water to replace steam generator secondary inventory, lost as the plant cools to MODE 4 conditions.

During heatup, the turbine driven AFW pump is only required to be made OPERABLE prior to making the reactor critical. It is required to be OPERABLE during subsequent MODE 1, 2, and 3 operation. This allowance is needed to provide sufficient steam pressure to perform turbine and pump testing.

BASES

APPLICABILITY (continued)

In MODE 4, the AFW System may be used for heat removal via the steam generator.

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.1

Condition A is applicable whenever one or more AFW trains is inoperable, in MODE 1, 2, or 3. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the assumption that at least 100% of the required AFW flow (that assumed in the safety analyses) is available to each steam generator. If the flow available to either steam generator is less than 100% of the required AFW flow, or if less than two AFW pumps are OPERABLE, Condition B must also be entered. In addition, if the combined flow available to both steam generators is less than 100% of the required AFW flow, Condition C must be entered as well.

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

The AFW system can provide one hundred percent of the required AFW flow to each steam generator following the occurrence of any single active failure. Therefore, the AFW function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurrence of an additional failure cannot be guaranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

BASES

ACTIONS (continued)

B.1 and B.2

Condition B is applicable: 1) when the Required Actions of Condition A cannot be completed within the required Completion Time, 2) when the flow available to either steam generator is less than 100% of the required AFW flow, or 3) when less than two AFW pumps are OPERABLE. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If the combined flow available to both steam generators is less than 100% of the required AFW flow, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

Continued plant operation is not allowed if the available AFW flow to either steam generator is less than the required flow, because adequate AFW flow cannot be assured following a main steam line break affecting that steam generator (consider the case where the break occurs in the AFW piping). Therefore, if 1) the inoperable AFW trains cannot be restored to OPERABLE status within the required Completion Time of Condition A, or 2) the flow available to either steam generator is less than 100% of the required AFW flow, or 3) less than two AFW pumps are OPERABLE in MODES 1, 2, and 3, the plant must be placed in a MODE in which the LCO does not apply (except as noted in Condition C). To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1

Condition C is applicable if the combined flow available to both steam generators is less than 100% of the required AFW flow; Condition A is applicable whenever one or more trains is inoperable; and Condition B is applicable when the flow available to either steam generator is less than 100% of the required AFW flow, or when less than two AFW pumps are OPERABLE. Therefore, when Condition C is applicable, Conditions A and B are also applicable. Being in Conditions A, B, and C concurrently maintains the Completion Time clocks for instances where equipment repair allows exit from Condition C while the plant is still within the applicable conditions of the LCO.

One hundred percent AFW flow (that assumed in the safety analyses) can be provided by any one OPERABLE AFW pump and an OPERABLE flow path to each steam generator.

Required Action C.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least 100% of the required AFW flow is available. In this condition, there may be inadequate AFW flow available to remove decay heat and allow a stable plant shutdown.

With less than 100% of the required AFW flow available (ie. less than the AFW flow assumed in the safety analyses, while in MODES 1, 2, and 3, or less than the required AFW train OPERABLE while in MODE 4 with a steam generator relied upon for heat removal), the plant is in a seriously degraded Condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least 100% of the required AFW flow available. LCO 3.0.3 is not applicable, as it could force the plant into a less safe condition.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for the required manual, power operated, and automatic valves in the AFW water and steam supply flow path 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 Surveillance 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.

This test need not be performed for the steam driven AFW pump for MODE 4 operation.

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

Verifying that each required AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of pump performance required by Section XI of the ASME Code (Ref. 2). 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.

This SR is modified by a Note indicating that this SR for the turbine driven AFW pump does not have to be met in MODE 3 when steam pressure is below 800 psig. This is because there is insufficient steam pressure and pump discharge pressure to allow the turbine driven pump to reach the normal test conditions.

Performance of inservice testing, discussed in the ASME Code, Section XI (Ref. 2), at 3 month intervals satisfies this requirement.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

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, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Specific signals (e.g., AFAS) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note which states the SR is only required to be met in MODES 1, 2, and 3 when AFW is not in operation. With AFW in operation, the required trains are already aligned with the flow control valves in manual control.

SR 3.7.5.4

This SR ensures that the AFW pumps will start in the event of any accident or transient that generates an AFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. Specific signals (e.g., AFAS, handswitch) are tested under Section 3.3, "Instrumentation."

This test need not be performed for the steam driven AFW pump for MODE 4 operation.

The 18 month Frequency is acceptable, based on the design reliability and operating experience of the equipment.

This SR is modified by a Note. The Note states that the SR is only required to be met in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required.

REFERENCES

1. FSAR, Section 9.7
2. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.
3. Palisades Design Basis Document 1.03, Auxiliary Feedwater System, Section 3.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage and Supply

BASES

BACKGROUND

The Condensate Storage and Supply provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Primary Coolant System (PCS). The Condensate Storage Tank (CST) and the Primary Makeup Storage Tank (T-81) provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5, "Auxiliary Feedwater (AFW) System"). Three AFW pumps take a suction from a common line from the CST. T-81 provides makeup to the CST either by use of a pump or by gravity flow. Backup sources from the Service Water System (SWS) and Fire Water System an provide additional water supply to the AFW pump suctions if the normal source is lost. SWS provides an emergency source to AFW pump P-8C, and the Fire Water System provides an emergency source to AFW pumps P-8A and P-8B. The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the turbine bypass valve. 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 PCS, it is designed to withstand earthquakes. The tornado protected supply is provided by the SWS and Fire Water System. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply.

A description of the Condensate Storage and Supply is found in the FSAR, Section 9.7 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES	The Condensate Storage and Supply provides condensate to remove decay heat and to cool down the plant following all events in the accident analysis, discussed in the FSAR, Chapters 5 and 14. For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs followed by a cooldown to Shutdown Cooling (SDC) entry conditions at the design cooldown rate. The Condensate Storage and Supply satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	To satisfy accident analysis assumptions, the CST and T-81 must contain sufficient cooling water to remove decay heat for 8 hours following a reactor trip from 102% RTP. This amount of time allows for cool down of the PCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this the CST and T-81 must retain sufficient water to ensure adequate net positive suction head for the AFW pumps, and makeup for steaming required to remove decay heat. The combined CST and T-81 level required is a usable volume of at least 100,000 gallons, which is based on holding the plant in MODE 3 for 4 hours, followed by a cooldown to SDC entry conditions at approximately 75°F per hour. This basis was established by the Systematic Evaluation Program. OPERABILITY of the Condensate Storage and Supply System is determined by maintaining the combined tank levels at or above the minimum required volume.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the Condensate Storage and Supply is required to be OPERABLE. In MODES 5 and 6, the Condensate Storage and Supply is not required because the AFW System is not required.

BASES

ACTIONS

A.1 and A.2

If the condensate volume is not within the limit, the OPERABILITY of the backup water supplies must be verified by administrative means within 4 hours and once every 12 hours thereafter.

OPERABILITY of the backup feedwater supplies must include verification of the OPERABILITY of flow paths from the Fire Water System and SWS to the AFW pumps, and availability of the water in the backup supplies. The Condensate Storage and Supply volume must be returned to OPERABLE status within 7 days, as the backup supplies may be performing this function in addition to their normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the Fire Water System and SWS. Additionally, verifying the backup water supplies every 12 hours is adequate to ensure the backup water supplies continue to be available. The 7 day Completion Time is reasonable, based on OPERABLE backup water supplies being available, and the low probability of an event requiring the use of the water from the CST and T-81 occurring during this period.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.7.6 A.1 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 12 hours" interval may utilize the 25% SR 3.0.2 extension.

B.1 and B.2

If the condensate volume cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the combination of CST and T-81 contain the required useable volume of cooling water. (This volume ≥ 100,000 gallons.) The 12 hour Frequency is based on operating experience, and the need for operator awareness of plant evolutions that may affect the Condensate Storage and Supply 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 and T-81 level deviations.

REFERENCES

1. FSAR, Section 9.7

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System (SWS), and thus to the environment.

The isolation of the CCW to components or systems may render those components inoperable but does not affect the OPERABILITY of the CCW System.

The CCW System consists of three pumps connected in parallel to common suction and discharge headers. Any single CCW pump can provide one hundred percent of the required CCW post accident cooling capability. The discharge header splits into two parallel heat exchangers and then combines again into a common distribution header which supplies various heat loads. A common surge tank provides the necessary net positive suction head for the CCW pumps and a surge volume for the system. A train of CCW is considered to be that equipment electrically connected to a common safety bus necessary to transfer heat acquired from the various heat loads to the SWS. There are two CCW trains, each associated with a Safeguards Electrical Distribution Train which are described in Specification 3.8.9, "Distribution Systems - Operating."

1. The CCW train associated with the Left Safeguards Electrical Distribution Train consists of two CCW pumps (P-52A, P-52C), CCW heat exchanger E-54B, the CCW surge tank (T-3), associated piping, valves receiving an actuation signal from the left train (eg. CV-0911, CV-0938, & CV-0946), and controls for that equipment to perform their safety function.

2. The CCW train associated with the Right Safeguards Electrical Distribution Train consists of one CCW pump (P-52B), CCW heat exchanger E-54A, the CCW surge tank (T-3), associated piping, CCW control valves receiving an actuation signal from the right train (eg. CV-0937, CV-0940, & CV-0945, and controls for that equipment to perform their safety function.

BASES

BACKGROUND (continued)

3. The CCW system piping, CCW Surge Tank (T-3), and those CCW control valves which receive actuation signals from both right and left trains (eg. CV-0910, CV-0913, CV-0944, CV-0944A, CV-0950, & CV-0977B) controls for that equipment are common to both trains.

CCW system components receive three automatic actuation signals, a Safety Injection Signal (SIS), a Recirculation Actuation Signal (RAS), or a Containment High Pressure (CHP) signal:

1. SIS starts the CCW pumps, isolates non-essential CCW loads outside the containment, opens the CCW inlet valves to the Shutdown Heat Exchangers (SDHXs), and sends an open signal to the engineered safeguards pump cooler CCW inlet valves (which are normally open).
2. RAS sends an open signal to the CCW heat exchanger CCW inlet valves (which are normally open).
3. CHP isolates the CCW loads inside the containment.

The CCW System cools three groups of loads which are described in the FSAR (Ref. 1). The major loads are:

1. Safety related loads outside the containment,
   Shutdown Cooling Heat Exchangers
   Engineered Safeguards Pump Coolers
   Charging Pump Oil Coolers
2. Non-safety related loads outside the Containment, and
   Spent Fuel Cooling Heat Exchangers
   Waste Gas Compressors
   Rad Waste Evaporators
3. Non-safety related loads inside the Containment.
   Letdown Heat Exchanger
   Shield Cooling Heat Exchangers
   Primary Coolant Pump Leakoff and Oil Coolers
   CRDM Seal Coolers

Each of these groups of loads can be cooled by the flow from one CCW pump. During normal operation, when full flow is not being provided to the Shutdown Cooling and Letdown Heat Exchangers, one CCW pump can provide the required flow for all three groups of loads. Two pumps are normally operated to provide additional system flow and thermal stability.

BASES

BACKGROUND (continued)

During post accident conditions, with all CCW and related system components OPERABLE, one hundred percent of the required CCW post accident cooling capability can be provided by any one CCW pump with sufficient flow margin to allow manually restoring CCW flow to the Spent Fuel Pool Cooling Heat Exchangers. If CCW or related systems have components out of service, additional CCW pumps may be required to provide the required cooling capability.

For post accident cooling, the Engineered Safety Features signals reposition several valves to maximize containment cooling and conserve CCW flow. Initially, a safety injection signal will start the CCW pumps, and open the large CCW inlet valves to the Shutdown Cooling Heat Exchangers (CCW cools the Shutdown Cooling Heat Exchangers, which cool the containment spray flow). A safety injection signal will also isolate the non-safety related CCW loads outside the containment. A Containment High Pressure signal will isolate the non-safety related CCW loads inside the containment. The occurance of these automatic actions will provide the required CCW post accident cooling capability while limiting the CCW flow requirement to that which can be provided by one CCW pump.

The safety analyses assume that both CCW heat exchangers are available. To assure that both heat exchangers will be available even with a single active failure, the CCW inlet valves to the CCW heat exchangers are maintained in the full open position during plant operation.

If the CCW side (shell side) of either CCW Heat Exchanger is out of service, 100% of the required CCW post accident cooling capability cannot be assured. If the SWS side (tube side) of either CCW Heat Exchanger is out of service, 100% of the required CCW post accident cooling capability can be provided, if other equipment outages are limited. One hundred percent of the post accident CCW cooling can be provided with the SWS side of one CCW Heat Exchanger out of service if the following equipment is OPERABLE: 3 safety related Containment Air Coolers, 2 Containment Spray Pumps, 3 CCW pumps, 2 SWS pumps, and both Shutdown Cooling Heat Exchangers, and if

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, <u>and</u>

2. One CCW header isolation valve for the non-safety related loads outside the containment, CV-0944, CV-0944A, or CV-0977B, is OPERABLE.

BASES

BACKGROUND (continued)

One hundred percent of the required CCW post accident cooling capability can be provided by one CCW pump if both CCW heat exchangers are available and CCW flow to both non-safety related flow paths can be isolated. (Since one pump can supply the safety-related loads and the Spent Fuel Pool Cooling Heat Exchangers, isolation capability for that heat exchanger is not necessary.) The necessary isolation of each non-safety related CCW flow path may be accomplished by any one of three valves.

1. The capability to isolate CCW flow to the non-safety related loads in the containment requires one CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, to be OPERABLE.

2. The capability to isolate CCW flow to the non-safety related loads outside the containment requires one CCW header isolation valve in the non-safety related CCW header outside the containment, CV-0944, CV-0944A, or CV-0977B, to be OPERABLE.

One hundred percent of the required CCW post accident cooling capability can be provided by two CCW pumps if both CCW heat exchangers are available and CCW flow to either non-safety related flow path can be isolated.

One hundred percent of the required CCW post accident cooling capability can be provided by three CCW pumps if both CCW heat exchangers are available, even with CCW flow being provided to both the safety-related loads and the non-safety related loads both inside and outside the containment.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.3 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchangers. This may utilize the SDC heat exchangers during a normal or post accident cooldown and shutdown in conjunction with the Containment Spray System during the recirculation phase following a LOCA.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train in conjunction with the SWS and a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat between 20 to 40 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 Primary Coolant System (PCS) by the safety injection pumps. Any single CCW pump can provide one hundred percent of the required CCW post accident cooling capability if both CCW heat exchangers are available.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power. The CCW System also functions to cool the plant from SDC entry conditions (T_{ave} < 300°F) to MODE 5 (T_{ave} < 200°F) during normal and post accident operations. The time required to cool from 300°F to 200°F is a function of the number of CCW and SDC trains operating. This assumes that the maximum Lake Michigan water temperature of LCO 3.7.9, "Ultimate Heat Sink (UHS)," occurs simultaneously with the maximum heat loads on the system.

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

BASES

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies. In the event of a DBA, one CCW train 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 CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

The CCW train associated with the Left Safeguards Electrical Distribution Train is considered OPERABLE when:

a. CCW pumps P-52A and P-52C are OPERABLE;

b. CCW Surge Tank T-3 and other common components are OPERABLE;

c. CCW heat exchanger E-54B is OPERABLE; and

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

The CCW train associated with the Right Safeguards Electrical Distribution Train is considered OPERABLE when:

a. CCW pump P-52B is OPERABLE;

b. CCW Surge tank T-3 and other common components are OPERABLE;

c. CCW heat exchanger E-54A is OPERABLE; and

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

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

BASES

APPLICABILITY

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

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

ACTIONS

A.1

Condition A is applicable whenever one or more CCW trains is inoperable. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the assumption that at least 100% of the required CCW post accident cooling capability (that assumed in the safety analyses) is available. (If, however, less than 100% of the CCW post accident cooling is available, Condition C must also be entered.)

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

The CCW system can provide one hundred percent of the required post accident cooling capability following the occurance of any single active failure. Therefore, the CCW function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurance of an additional failure cannot be guaranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

B.1 and B.2

B is applicable when the Required Actions of Condition A cannot be completed within the required Completion Time. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If less than 100% of the post accident CCW cooling capability is available, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances

BASES

ACTIONS A.1 & A.2 (Continued)

where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

If the required CCW trains cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

C.1

Condition C is applicable with one or more trains inoperable when there is less than 100% of the required CCW post accident cooling capability available. Condition A is applicable whenever one or more trains is inoperable. Therefore, when this Condition is applicable, Condition A is also applicable. Being in Conditions A and C concurrently maintains both Completion Time clocks for instances where equipment repair restores 100% of the required CCW post accident cooling capability while the LCO is still applicable, allowing exit from Condition C (and LCO 3.0.3).

Any single CCW pump can provide one hundred percent of the required CCW post accident cooling capability. One hundred percent of the required CCW post accident cooling can also be provided despite the inoperability of one or more of those CCW valves closed by Safety Injection, which isolate cooling to non-essential loads, provided there are sufficient CCW pumps available to supply the additional flow.

The Component Cooling System cools three groups of loads:

1. Safety related loads outside the containment,
2. Non-safety related loads outside the Containment, and
3. Non-safety related loads inside the Containment.

As discussed in the Background section of these bases, each of these groups of loads can be cooled by the flow from one CCW pump.

One hundred percent of the required CCW post accident cooling capability can be provided by one CCW pump if both CCW heat exchangers are available and if:

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, and

BASES

ACTIONS C.1 (continued)

2. One CCW header isolation valve for the non-safety related loads outside the containment, CV-0944, CV-0944A, or CV-0977B, is OPERABLE.

One hundred percent of the required CCW post accident cooling capability can be provided by two CCW pumps if both CCW heat exchangers are available and if:

1. One CCW Containment Isolation Valve, CV-0910, CV-0911, or CV-0940, is OPERABLE, or

2. One CCW header isolation valve for the non-safety related loads outside the containment, CV-0944, CV-0944A, or CV-0977B, is OPERABLE.

One hundred percent of the required CCW post accident cooling capability can be provided by three CCW pumps if both CCW heat exchangers are available, even with CCW flow being provided to both the safety-related loads and the non-safety related loads inside and outside the containment.

With less than 100% of the required CCW post accident cooling capability available, the plant is in a condition outside the assumptions of the safety analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.7.1

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

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection, RAS) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.3

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation or a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The isolation of the SWS to components or systems may render those components inoperable but does not affect the OPERABILITY of the SWS System.

The SWS consists of three pumps connected in parallel taking suction from a common intake structure supplied by Lake Michigan. The discharge of the pumps flow into a common header before splitting into three headers (two critical headers for safety-related equipment and a single non-critical header for non safety-related equipment). The return piping from the three headers join into a common line and discharge to the cooling tower makeup basin. A train of SWS shall be that equipment electrically connected to a common safety bus necessary to remove heat from the various heat loads. There are two SWS trains, each associated with a Safeguards Electrical Train which are described in Specification 3.8.9, "Distribution Systems - Operating." The SWS train associated with the Left Safeguards Train consists of one SWS pump (P-7B), associated piping, valves, and controls for the equipment to perform their safety function. The SWS train associated with the Right Safeguards Train consists of two SWS pumps (P-7A, P-7C), associated piping, valves, and controls for the equipment to perform their safety function. The pumps and valves are remote manually aligned, except in the unlikely event of a Loss Of Coolant Accident (LOCA).

SWS components receive three automatic actuation signals, a Safety Injection Signal (SIS), a Recirculation Actuation Signal (RAS), or a Diesel Generator (DG) start signal:

1. SIS starts the SWS pumps, isolates the non-critical service water header, and realigns the Containment Air Cooler (CAC) service water valves to the post accident cooling configuration.

BASES

BACKGROUND (continued)

2. RAS realigns the CCW heat exchanger service water outlet valves for maximum cooling.

3. A DG start signal opens the DG lube oil and jacket water cooler inlet valves.

The DG which powers two SWS pumps (P-7A, P-7C), also powers the fans associated with VHX-1, VHX-2, and VHX-3 (V-1A, V-2A and V-3A). This is necessary because if reliance for containment cooling is placed on CACs, at least two service water pumps must be OPERABLE to provide the necessary service water flow to assure OPERABILITY of the CACs. The Service Water System cools three groups of loads. The SWS loads are described in the FSAR (Ref. 1), the major loads are:

1. Critical loads inside the Containment,
   Containment Air Coolers VHX-1, VHX-2, VHX-3, (and VHX-4).

2. Critical loads outside the Containment, and
   Diesel Generators 1-1 and 1-2
   Component Cooling Heat Exchangers E-54A and E-54B
   Engineered Safeguards Room Coolers VHX-27A and VHX-27B
   Control Room HVAC Coolers VC-10 and VC 11
   Instrument Air Compressors C-2A and C-2C

3. Non-critical loads in the Turbine Building

Each of these groups of loads can be cooled by the flow from one SWS pump. During normal operation, when SWS flow from the CACs and CCW heat exchangers is throttled by temperature control valves, two SWS pumps can provide the required flow for all three groups of loads.

During post accident conditions, with all other SWS and related system components OPERABLE, one hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump. If SWS or related systems have components out of service, additional SWS pumps may be required to provide the required cooling capability.

For post accident cooling, the Engineered Safety Features signals reposition several valves to maximize containment cooling and conserve SWS flow. Initially, a safety injection signal will start the SWS pumps, realign the SWS valves for the CACs (which cool the containment atmosphere), and close the non-critical SWS header isolation valve.

BASES

BACKGROUND (continued)

Subsequently, if the Safety Injection Refueling Water Tank has been emptied, a RAS will realign the SWS outlet valves on the CCW heat exchangers (CCW cools the Shutdown Cooling Heat Exchangers, which cool the containment spray flow). The occurance of these automatic actions will provide the one hundred percent of the required post accident SWS cooling capability while limiting the SWS flow requirement to that which can be provided by two SWS pumps.

If the Containment Air Coolers are not needed for post accident containment cooling. SWS flow to the containment may then be isolated, further reducing the required SWS post accident cooling capability to that which can be provided by one SWS pump.

One hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump if SWS flow <u>both</u> to the non-critical header and to the critical loads inside the containment are capable of being isolated.

1. The capability to isolate SWS flow to the non-critical SWS header requires its isolation valve, CV-1359, to be OPERABLE.

2. The allowance to isolate SWS flow to the containment requires the ability to provide post accident containment cooling without reliance on CACs.

   The capability to isolate SWS flow to the containment requires one SWS Containment Isolation Valve, CV-0824 or CV-0847, to be OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by any two SWS pumps if SWS flow <u>either</u> to the non-critical header or to the critical loads inside the containment are capable of being isolated.

One hundred percent of the required SWS post accident cooling capability can be provided by three SWS pumps even with SWS flow being provided to both the CACs and the Non-critical SWS header.

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the FSAR, Section 9.1 (Ref. 1). The principal safety related functions of the SWS is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System and the removal of heat from the containment atmosphere via the CACs.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system (containment spray, CACs, or a combination), removing core decay heat between 20 to 40 minutes following a design basis LOCA. 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 Primary Coolant System by the safety injection pumps. The SWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.

The SWS, in conjunction with the CCW System, also cools the plant from Shutdown Cooling (SDC) entry Condition, as discussed in the FSAR, Section 6.1 (Ref. 2) to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of CCW and SDC System trains that are operating. This assumes that the maximum Lake Michigan water temperature of LCO 3.7.9, "Ultimate Heat Sink (UHS)," occurs simultaneously with maximum heat loads on the system.

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

LCO

Two SWS trains 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.

The SWS train associated with the Left Safeguard Electrical Distribution Train is considered OPERABLE when:

a. SWS pump P-7B is OPERABLE; and
b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

The SWS train associated with the Right Safeguards Electrical Distribution Train is OPERABLE when:

a. SWS pumps P-7A and P-7C are OPERABLE; and
b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

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

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS System is a normally operating system, which is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES. In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

A.1

Condition A is applicable whenever one or more SWS trains is inoperable. Action A.1 requires restoration of both trains to OPERABLE status within 72 hours. The 72 hour Completion Time is based on the assumption that at least 100% of the required SWS post accident cooling capability (that assumed in the safety analyses) is available. (If, however, less than 100% of the SWS post accident cooling is available, Condition C must also be entered.)

Mechanical system LCOs typically provide a 72 hour Completion Time under conditions when a required system can perform its required safety function, but may not be able to do so assuming an additional failure. When operating in accordance with the Required Actions of an LCO Condition, it is not necessary to be able to cope with an additional single failure.

The SWS system can provide one hundred percent of the required post accident cooling capability following the occurrence of any single active failure. Therefore, the SWS function can be met during conditions when those components which could be deactivated by a single active failure are known to be inoperable. Under that condition, however, the ability to provide the function after the occurrence of an additional failure cannot be guaranteed. Therefore, continued operation with one or more trains inoperable is allowed only for a limited time.

B.1 and B.2

Condition B is applicable when the Required Actions of Condition A cannot be completed within the required Completion Time. Condition A is applicable whenever one or more trains is inoperable. Therefore, when Condition B is applicable, Condition A is also applicable. (If less than 100% of the post accident SWS cooling capability is available, Condition C must be entered as well.) Being in Conditions A and B concurrently maintains both Completion Time clocks for instances where equipment repair allows exit from Condition B while the plant is still within the applicable conditions of the LCO.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable SWS trains cannot be restored to OPERABLE status within the associated required Completion Time of Condition A, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition C is applicable with one or more trains inoperable when there is less than 100% of the required SWS post accident cooling capability available. Condition A is applicable whenever one or more trains is inoperable. Therefore, when this Condition is applicable, Condition A is also applicable. Being in Conditions A and C concurrently maintains both Completion Time clocks for instances where equipment repair restores 100% of the required SWS post accident cooling capability while the LCO is still applicable, allowing exit from Condition C (and LCO 3.0.3).

The Service Water System cools three groups of loads:

1. Critical loads inside the Containment,
2. Critical loads outside the Containment, and
3. Non-critical loads in the Turbine Building.

As discussed in the Background section of these bases, each of these groups of loads can be cooled by the flow from one SWS pump.

One hundred percent of the required SWS post accident cooling capability can be provided by any one SWS pump if:

1. The non-critical SWS header isolation valve, CV-1359, is OPERABLE, and

2. Plant conditions allow adequate containment cooling to be provided without reliance on CACs and one SWS Containment Isolation Valve, CV-0824 or CV-0847, is OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by any two SWS pumps if:

BASES

ACTIONS (continued)

C.1

1. The non-critical SWS header isolation valve, CV-1359, is OPERABLE, or

2. Plant conditions allow adequate containment cooling to be provided without reliance on CACs and one SWS Containment Isolation Valve, CV-0824 or CV-0847, is OPERABLE.

One hundred percent of the required SWS post accident cooling capability can be provided by three SWS pumps even with SWS flow being provided to both the CACs and the Non-critical SWS header.

With less than 100% of the required SWS post accident cooling capability available, the plant is in a condition outside the assumptions of the safety analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

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

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. Specific signals (e.g., safety injection) are tested under Section 3.3, "Instrumentation." This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, and 3. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal in the "with standby power available" mode which tests the starting of the pumps by the SIS-X relays. The starting of the pumps by the sequencer is performed in Section 3.8, "Electrical Power Systems." This SR is modified by a Note which states this SR is not required to be met in MODE 4. The instrumentation providing the input signal is not required in MODE 4, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met in this MODE. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.1
2. FSAR, Section 6.1

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for process and operating heat from safety related components during a Design Basis Accident (DBA) or transient, as well as during normal operation. This is done utilizing the Service Water System (SWS).

The UHS has been defined as Lake Michigan. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that an adequate Net Positive Suction Head (NPSH) to the SWS pumps be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the system along with a list of components served can be found in FSAR, Section 9.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the plant is cooled down and placed on shutdown cooling. Maximum post accident heat load occurs between 20 to 40 minutes after a design basis Loss of Coolant Accident (LOCA). Near this time, the plant switches from injection to recirculation, and the containment cooling systems are required to remove the core decay heat.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The minimum water level of the UHS is based on the NPSH requirements for the SWS pumps. The NPSH calculation assumes a minimum water level of 4 feet above the bottom of the pump suction bell which corresponds to an elevation of 557.25 ft. Violation of the SWS pump submergence requirement should never become a factor unless the Lake Michigan water level falls below the top of the sluice gate opening which is at elevation 568.25 ft. Early warning of a falling intake water level is provided by the intake structure level alarm. The nominal lake level is approximately 580 ft mean sea level. The maximum water temperature of the UHS is based on conservative heat transfer analyses for the worst case LOCA. FSAR, Section 14.18 (Ref. 2) and Design Basis Document (DBD) 1.02 (Ref. 3) provide the details of the analysis which forms the basis for these operating limits. The assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case single active failure.

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

LCO

The UHS is required to be OPERABLE. The UHS is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the SWS to operate without the loss of NPSH, and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature shall not exceed 85°F and the level shall not fall below 568.25 ft above mean sea level during normal plant operation.

APPLICABILITY

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

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

BASES

ACTIONS	A.1 and A.2 If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.9.1 This SR verifies adequate cooling can be maintained. The level specified also ensures sufficient NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ 568.25 ft above mean sea level as measured within the boundaries of the intake structure. SR 3.7.9.2 This SR verifies that the SWS is available to provide adequate cooling for the maximum accident or normal design heat loads following a DBA. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the water temperature from the UHS is ≤ 85°F.
REFERENCES	1. FSAR, Section 9.1 2. FSAR, Section 14.18 3. Design Basis Document (DBD) 1.02, "Service Water System"

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Ventilation (CRV) Filtration

BASES

BACKGROUND

The CRV Filtration provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity.

The CRV Filtration consists of a common emergency intake which splits into two independent, redundant trains that recirculate and filter the control room air. The exhaust of each train exhausts into a common supply plenum. Each train consists of a prefilter, a heater, a High Efficiency Particulate Air (HEPA) filter, two banks of activated charcoal adsorbers for removal of gaseous activity (principally iodine), a second HEPA filter, and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and to back up the main HEPA filter bank if it fails.

The CRV Filtration is an emergency system, part of which may also operate during normal plant operations in the standby mode of operation. Upon manual initiation or receipt of a containment high pressure or containment high radiation signal, normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the filter trains of the system. The prefilters remove any large particles in the air. Continuous operation of each train for at least 10 hours per month with the heaters on reduces moisture buildup on the HEPA filters and adsorbers. The heater is important to the effectiveness of the charcoal adsorbers.

Actuation of the system to the emergency mode of operation closes the normal unfiltered outside air intake and unfiltered exhaust dampers, opens the emergency air intake, and aligns the system for recirculation of control room air through the redundant trains of HEPA and charcoal filters. The emergency mode initiates pressurization and filtered ventilation of the air supply to the control room.

Outside air is filtered, and then added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building.

BASES

BACKGROUND (continued)	A single train will pressurize the control room to at least 0.125 inches water gauge relative to the south hallway outside the Control Room Viewing Galley, and provides an air exchange rate in excess of 25% per hour. The CRV Filtration operation in maintaining the control room habitable is discussed in the FSAR, Section 9.8 (Ref. 1). Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across one filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRV Filtration is designed in accordance with Seismic Category I requirements. The CRV Filtration is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.
APPLICABLE SAFETY ANALYSES	The CRV Filtration components are arranged in redundant safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CRV Filtration provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis events discussed in the FSAR, Chapter 14 (Ref. 2). The worst case single active failure of a component of the CRV Filtration, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The CRV Filtration satisfies Criterion 3 of 10 CFR 50.36(c)(2).
LCO	Two independent and redundant trains of the CRV Filtration are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train. Total system failure could result in a control room operator receiving a dose in excess of 5 rem in the event of a large radioactive release.

BASES

LCO (continued)

The CRV Filtration is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both trains. A CRV Filtration train is considered OPERABLE when the associated:

a. Main recirculation fan and emergency filter fan are OPERABLE;

b. HEPA filters and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Required heater, ductwork, valves, and dampers are OPERABLE,

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors such that 0.125 inches water gauge positive pressure can be maintained in the emergency mode.

This LCO is modified by a Note allowing the control room boundary to be opened intermittently under administrative control. Since this Note modifies the LCO, no Condition entry is required when the control room boundary is opened under its provisions. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CRV Filtration must be OPERABLE to limit operator exposure during and following a DBA.

In MODES 5 and 6, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining CRV Filtration OPERABLE is not required in MODE 5 or 6, except for the following situations under which significant radioactive releases can be postulated:

a. During CORE ALTERATIONS;

b. During movement of irradiated fuel assemblies; and

c. During movement of a fuel cask in or over the SFP.

BASES

ACTIONS

A.1

With one CRV Filtration train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRV Filtration subsystem is adequate to perform control room radiation protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRV Filtration train could result in loss of CRV Filtration function. The 7 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.

B.1 and B.2

If the control room boundary is inoperable in MODE 1, 2, 3, or 4, the CRV Filtration trains cannot perform their functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of General Design Criterion 19) shall be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, or smoke, and to ensure control room physical security. Preplanned measures shall be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of an analyzed event requiring control room isolation occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a reasonable time to diagnose and repair most problems occurring with the control room boundary.

C.1 and C.2

If the inoperable CRV Filtration train or control room boundary cannot be restored to OPERABLE status within the required Completion Time of Condition A or B in MODE 1, 2, 3, or 4, the plant must be placed in a MODE that minimizes the accident risk. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS

D.1, D.2.1, D.2.2, and D.2.3

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, during movement of a fuel cask in or over the SFP, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CRV Filtration train must be immediately placed in the emergency mode of operation. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure will be readily detected.

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

E.1, E.2, and E.3

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, with two CRV Filtration trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

F.1

If both CRV Filtration trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), the CRV Filtration may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

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 train once every month provides an adequate check on this system.

Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each train must be operated for ≥ 10 continuous hours with the associated heater, VHX-26A or VHX-26B, energized. The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.

SR 3.7.10.2

This SR verifies that the required CRV Filtration testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRV Filtration filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3) as described in the VFTP. 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 3.7.10.3

This SR verifies each CRV Filtration train starts and operates on an actual or simulated actuation signal. Specific signals (e.g., containment high pressure, containment high radiation) are tested under Section 3.3, "Instrumentation." This SR is modified by a Note which states this SR is only required to be met in MODES 1, 2, 3 and 4 and during movement of irradiate fuel assemblies in containment. The instrumentation providing the input signal is not required in these plant conditions, therefore, to keep consistency with Section 3.3, "Instrumentation," the SR is not required to be met. The Frequency of 18 months is consistent with that specified in Reference 3.

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CRV Filtration. During the emergency mode of operation, the CRV Filtration is designed to pressurize the control room ≥ 0.125 inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CRV Filtration is designed to maintain this positive pressure with one train at an emergency ventilation flow rate of ≥ 3040 cfm and ≤ 3520 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.4 (Ref. 4).

REFERENCES

1. FSAR, Section 9.8
2. FSAR, Chapter 14
3. Regulatory Guide 1.52 (Rev. 2)
4. NUREG-0800, Section 6.4, Rev. 2, July 1981

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Ventilation (CRV) Cooling System

BASES

BACKGROUND

The CRV Cooling provides temperature control for the control room during normal and emergency conditions.

The CRV Cooling consists of two independent, redundant trains, which exhaust into a common supply plenum that provide cooling and heating of recirculated control room air. In the emergency mode, the two trains are supplied by a common emergency intake which splits into the two trains. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control. The CRV Cooling is a subsystem providing air temperature control for the control room.

The CRV Cooling is an emergency system, parts of which may also operate during normal plant operations. A single train will provide the required temperature control to maintain the control room at 90°F or below. The CRV Cooling operation to maintain the control room temperature is discussed in the FSAR, Section 9.8 (Ref. 1).

The control room ventilation emergency mode of operation is actuated either by a containment high radiation signal or a containment high pressure signal, or manually from the control room. During emergency mode operation, the air handling units and the charcoal filter units of both Train A and Train B are actuated automatically. The CRV Cooling refrigerant Condensing Units VC-10 and VC-11 shut down and are manually restarted by the operator when their operation is required for control room cooling. In addition, since immediate operation of the CRV Cooling System is not necessary, other manual operations may be required to initiate control room cooling, depending on the configuration of the system upon initiation of the emergency mode signal.

BASES

APPLICABLE SAFETY ANALYSES

The design basis of the CRV Cooling is to maintain temperature of the control room environment throughout 30 days of continuous occupancy.

The CRV Cooling components are arranged in redundant safety related trains. During normal and emergency operation, the CRV Cooling maintains the temperature at 90°F or below, as required by LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." A single active failure of a component of the CRV Cooling, 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 CRV Cooling is designed in accordance with Seismic Category I requirements. The CRV Cooling 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 CRV Cooling satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Two independent and redundant trains of the CRV Cooling are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident. In addition, since immediate operation of the CRV Cooling System is not necessary, other manual operations may be required to initiate control room cooling, depending on the configuration of the system upon initiation of the emergency mode signal.

The CRV Cooling is considered OPERABLE when the individual components that are necessary to maintain the control room temperature are OPERABLE in both trains. These components include the condensing units, fans, and associated temperature control instrumentation. In addition, the CRV Cooling must be OPERABLE to the extent that air circulation can be maintained.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the CRV Cooling must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY requirements following isolation of the control room.

In MODES 5 and 6, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining CRV Cooling OPERABLE is not required in MODE 5 or 6, except for the following situations under which significant radioactive releases can be postulated:

a. During CORE ALTERATIONS;

b. During movement of irradiated fuel assemblies; and

c. During movement of a fuel cask in or over the SFP.

ACTIONS

A.1

With one CRV Cooling train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRV Cooling 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.

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes the accident risk. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

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

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CRV Cooling train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the plant in a condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

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

During CORE ALTERATIONS, during movement of irradiated fuel assemblies, or during movement of a fuel cask in or over the SFP, with two CRV Cooling trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the plant in a Condition that minimizes the accident risk. This does not preclude the movement of fuel assemblies or a fuel cask to a safe position.

E.1

If both CRV Cooling trains are inoperable in MODE 1, 2, 3, or 4, the CRV Cooling may not be capable of performing the intended function and the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. An 18 month Frequency is appropriate, since significant degradation of the CRV Cooling is slow and is not expected over this time period.
REFERENCES	1. FSAR, Section 9.8

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Handling Area Ventilation System

BASES

BACKGROUND

The Fuel Handling Area Ventilation System filters airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident. The fuel handling area is served by two separate subsystems one being part of the original plant design, and the other being added as part of the Auxiliary Building Addition.

The original plant design consists of a supply plenum and an exhaust plenum including associated ductwork, dampers, and instrumentation. The supply plenum contains one prefilter, two heating coils, and one supply fan. The exhaust plenum contains two filter banks (normal and emergency) configured in a parallel flow arrangement, and two independent exhaust fans which draw air from a common duct. The "normal filter bank" contains a prefilter and a High Efficiency Particulate Air (HEPA) filter. The "emergency filter bank" contains a prefilter, HEPA filter, and an activated charcoal filter.

The Auxiliary Building Addition, which was added to serve the spaces at the north end of the spent fuel pool, also consist of a supply plenum and exhaust plenum. The supply plenum is configured similar to the supply plenum provided in the original plant design and includes one prefilter, two heating coils, and one supply fan. The exhaust plenum is different from the original plant design in that it only contains one filter bank consisting of a prefilter and HEPA filter, and two common exhaust fans.

During normal plant operations, the Fuel Handling Area Ventilation System supplies filtered and heated (as needed) outside air to the fuel handling area. The exhaust fans draw air from the fuel handling area through the normally aligned prefilters and HEPA filters and discharge it to the unit stack by way of the main ventilation exhaust plenum.

BASES

BACKGROUND (continued)

During plant evolutions when the possibility for a fuel handling accident or fuel cask drop accident exist, the Fuel Handling Area Ventilation System is configured such that all fans are stopped except one exhaust fan in the original plant subsystem aligned to the "emergency filter bank." The "normal filter bank" in the original plant design is isolated by closing its associated inlet damper. Thus, in the event of a fuel handling accident, the fuel handling area atmosphere will be filtered for the removal of airborne fission products prior to being discharged to the outside environment.

The Fuel Handling Area Ventilation System is discussed in the FSAR, Sections 9.8, 14.11 and 14.19 (Refs. 1, 2, and 3) because it may be used for normal, as well as post-accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The Fuel Handling Area Ventilation System is designed to mitigate the consequences of a fuel handling accident or fuel cask drop accident by limiting the amount of airborne radioactive material discharged to the outside atmosphere.

The results and major assumptions used in the analysis of the fuel handling accident are presented in FSAR Section 14.19. For the purpose of defining the upper limit of the radiological consequences of a fuel handling accident, it is assumed that a fuel bundle is dropped during fuel handling activities and all the fuel rods in the equivalent of an entire assembly (216) fail. The bounding fuel handling accident is assumed to occur in containment two days after shutdown. No containment isolation is assumed to occur. As such, the released fission products escape to the environment with no credit for filtration. The results of this analysis have shown that the offsite doses resulting from this event are within the guideline of 10 CFR 100. In the event a fuel handling accident were to occur in the fuel handling area, the radioactive release would pass through the "emergency filter bank" significantly reducing the amount of radioactive material released to the environment. Thus, the consequences of a fuel handling accident in the fuel handling area are deemed acceptable with or without the "emergency filter bank" in operation since they are no more severe than the consequences of a fuel handling accident in containment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The results and major assumptions used in the analysis of the fuel cask drop accident are presented in FSAR Section 14.11. For the purpose of defining the upper limit of the radiological consequences of a fuel cask drop accident, it is assumed that all 73 fuel assemblies in a 7 x 11 Westinghouse spent fuel pool rack with a minimum decay of 30 days are damaged and release their fuel rod gap inventories. Three fuel cask drop scenarios were analyzed to encompass all fuel cask drop events. They are:

1. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. All isolatable unfiltered leak path are assumed to be isolated prior to event initiation.

2. A fuel cask drop onto 30 day decayed fuel with the Fuel Handling Area Ventilation System aligned for emergency filtration with a conservative amount of unfiltered leakage. This scenario determined the maximum amount of non-isolatable unfiltered leakage that can exist and still meet offsite dose limits. This scenario also assumes isolation of isolable leak paths prior to event initiation.

3. A fuel cask drop onto 90 day decayed fuel without the Fuel Handling Area Ventilation System aligned for emergency filtration. This scenario needs no assumptions as to unfiltered leakage or post-accident unfiltered leak path isolation times since all radiation is assumed to be released unfiltered from the fuel handling area.

The results of the analysis show that the radiological consequences of a fuel cask drop in the spent fuel pool meet the acceptance criteria of Regulatory Guide 1.25 (Ref. 4) and NUREG-0800 Section 15.7.5 (Ref. 5) for all scenarios. In addition, the dose from all scenarios are less than 25% of the dose guidelines in 10 CFR 100. For scenario 2, the analysis shows that a maximum of 20% charcoal filter bypass from non-isolatable leak paths can be accommodated while still meeting 25% of the 10 CFR 100 guidelines.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Filtration of the fuel handling area atmosphere following a fuel handling accident is not necessary to maintain the offsite doses within the guidelines of 10 CFR 100. Thus, a total system failure would not impact the margin of safety as described in the safety analysis. However, analysis has shown that post-accident filtration by the Fuel Handling Area Ventilation System provides significant reduction in offsite doses by limiting the release of airborne radioactivity. Therefore, for the fuel handling accident, the Fuel Handling Area Ventilation System satisfies Criterion 4 of 10 CFR 50.36(c)(2).

Filtration of the fuel handling area atmosphere following a fuel cask drop on irradiated fuel assemblies with < 90 days decay is required to maintain the offsite doses within the guidelines of 10 CFR 100. Therefore, for the fuel cask drop accident, the Fuel Handling Area Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the Fuel Handling Area Ventilation System ensures filtration of the fuel handling area atmosphere is immediately available in the event of a fuel handling accident, or a fuel cask drop accident. As such, the LCO requires the Fuel Handling Area Ventilation System to be OPERABLE with one fuel handling area exhaust fan aligned to the "emergency filter bank" and in operation.

The Fuel Handling Area Ventilation System is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE. The Fuel Handling Area Ventilation System is considered OPERABLE when:

a. One exhaust fan is aligned to the "emergency filter bank" and in operation to ensure the air discharged to the main ventilation exhaust plenum has been filtered. Operation of only one fuel handling area exhaust fan ensures the design flow rate of the "emergency filter bank" is not exceeded.

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

c. Ductwork and dampers are OPERABLE, and air circulation can be maintained. Inclusive to the requirement to align the "emergency filter bank" is that the "normal filter bank" is isolated by its associated inlet damper to prevent the release of unfiltered air.

BASES

APPLICABILITY

The Fuel Handling Area Ventilation System must be OPERABLE, aligned, and in operation whenever the potential exists for an accident that results in the release of radioactive material to the fuel handling area atmosphere that could exceed previously approved offsite dose limits if released unfiltered to the outside atmosphere. As such, the Fuel Handling Area Ventilation System is required; during movement of irradiated fuel assemblies in the fuel handling building when irradiated fuel assemblies with < 30 days decay time are in the fuel handling building; during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in containment when irradiated fuel assemblies with < 30 days decay time are in the containment with the equipment hatch open, and during movement of a fuel cask in or over the spent fuel pool when irradiated fuel assemblies with < 90 days decay time fuel handling building.

The requirement for the Fuel Handling Area Ventilation System does not apply during movement of irradiated fuel assemblies or CORE ALTERATIONS when all irradiated fuel assemblies in the fuel handling building, or all irradiated fuel assemblies in the containment with the equipment hatch open, have decayed for 30 days or greater since the dose consequences from a fuel handling accident would be of the same magnitude without the filters operating as the dose consequences would be with the filters operating and two days decay. In addition, the requirement for the Fuel Handling Area Ventilation System does not apply during fuel cask movement when all irradiated fuel assemblies in the fuel handling building have decayed 90 days or greater since the dose consequences remain less than 25% of the guidelines of 10 CFR 100.

ACTIONS

A.1, A.2, and A.3

If the Fuel Handling Area Ventilation System is not aligned to the "emergency filter bank", or one exhaust fan is not in operation, or the system is inoperable for any reason, action must be taken to place the unit in a condition in which the LCO does not apply. Therefore, activities involving the movement of irradiated fuel assemblies, CORE ALTERATIONS, and movement of a fuel cask in or over the spent fuel pool, must be suspended immediately to minimize the potential for a fuel handling accident.

The suspension of fuel movement, CORE ALTERATIONS, and fuel cask movement shall not preclude the completion of placing a fuel assembly, core component, or fuel cask in a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies the performance of Fuel Handling Area Ventilation System filter testing in accordance with the Ventilation Filter Testing Program. The Fuel Handling Area Ventilation System filter tests are in accordance with the Regulatory Guide 1.52 (Ref. 6) as described in Ventilation Filter Testing Program. The Ventilation Filter Testing Program 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 Ventilation Filter Testing Program.

SR 3.7.12.2

This SR verifies the Fuel Handling Area Ventilation System has not degraded and is operating as assumed in the safety analysis. The flow rate is periodically tested to verify proper function of the Fuel Handling Ventilation System. When aligned to the "emergency filter bank", the Fuel Handling Area Ventilation System is designed to reduce the amount of unfiltered leakage from the fuel handling building which, in the event of a fuel handling accident, lowers the dose at the site boundary to well within the guidelines of 10 CFR 100. The Fuel Handling Area Ventilation System is designed to lower the dose to these levels at a flow rate of $\geq$ 5840 cfm and $\leq$ 8760 cfm. The Frequency of 18 months is consistent with the test for filter performance and other filtration SRs.

BASES

REFERENCES

1. FSAR, Section 9.8
2. FSAR, Section 14.11
3. FSAR, Section 14.19
4. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Reactors.
5. NUREG-0800 Section 15.7.5, Spent Fuel Cask Drop Accidents.
6. Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorbtion Units of Light-Water-Cooled Nuclear Power Plants.

B 3.7 PLANT SYSTEMS

B 3.7.13 Engineered Safeguards Room Ventilation (ESRV) Dampers

BASES

BACKGROUND

The ESRV Dampers isolate the safeguards rooms by closing the inlet and exhaust plenum dampers on the initiation of a high radiation alarm from their respective airborne particulate monitor. This isolation lowers the offsite dose to well within 10 CFR 100 (Ref. 1) limits if a leak should occur. Typically, high radiation would only be expected due to excessive leakage during the recirculation phase of operation following a Loss of Coolant Accident (LOCA).

The ESRV Dampers consists of two trains. Each train consists of a supply plenum damper, a exhaust plenum damper, and associated piping, valves, and ductwork. Instrumentation which is addressed in LCO 3.3.10, "Engineered Safeguards Room Ventilation (ESRV) Instrumentation," also form part of the system, but is not addressed by this LCO. The Reactor Auxiliary Building Main Ventilation System provides normal cooling in conjunction with the engineered safeguards room coolers. Upon reciept of a high radiation signal, the ESRV Dampers are closed, isolating the affected safeguards room(s) from the rest of the auxiliary building ventilation system lowering the leakage to the environment from the auxiliary building.

The ESRV Dampers are discussed in the FSAR, Sections 7.4.5.2 and 14.22, and Design Basis Document (DBD) 1.07 (Refs. 2, 3, and 4, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the ESRV Dampers is established by the Maximum Hypothetical Accident (MHA). The system evaluation assumes a failure causes leakage into the engineered safeguards rooms, such as safety injection pump seal leakage, during the recirculation mode. In such a case, the system limits the radioactive release from the engineered safeguards rooms to well within 10 CFR 100 limits (Ref. 1). The analysis of the effects and consequences of a MHA is presented in Reference 3. The ESRV Dampers may also actuate following a small break LOCA, after the plant goes into the recirculation mode of long term cooling to mitigate releases of smaller leaks, such as from valve stem packing.

The ESRV Dampers satisfies Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

Two ESRV Damper trains are required to be OPERABLE to ensure that each engineered safeguards room isolates upon receipt of its respective high radiation alarm. Total system failure could result in the atmospheric release from the engineered safeguards rooms exceeding the required limits in the event of a Design Basis Accident (DBA).

An ESRV Damper train is considered OPERABLE when its associated instrumentation, ductwork, valves, and dampers are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESR-Damper trains are required to be OPERABLE consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).

In MODES 5 and 6, the ESRV Damper trains are not required to be OPERABLE, since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

Condition A addresses the failure of one or both ESRV Damper trains. Operation may continue as long as action is immediately initiated to isolate the affected engineered safeguards room. With the inlet and exhaust dampers closed, or if the inlet and outlet ventilation plenums are adequately sealed, the engineered safeguards room is isolated and the intended safety function is achieved, since the potential pathway for radioactivity to escape to the environment from the engineered safeguards room has been minimized.

The Completion Time for this Required Action is commensurate with the importance of maintaining the engineered safeguards room atmosphere isolated from the outside environment when the ECCS pumps are circulating primary coolant after an accident.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that each ESRV Damper train closes on an actual or simulated actuation signal. The 31 day Frequency is based on operating experience which has shown that these components usually pass the SR when tested at this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 100.11
2. FSAR, Section 7.4.5.2
3. FSAR, Section 14.22
4. Design Basis Document (DBD) 1.07, "Auxiliary Building HVAC Systems"

B 3.7 PLANT SYSTEMS

B 3.7.14 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 or cask drop 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 the FSAR, Section 9.11 (Ref. 1), and the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.4 (Ref. 2). The assumptions of fuel handling and fuel cask drop accidents are given in the FSAR, Section 14.19 and 14.11 (Refs. 3 and 4), respectively.
APPLICABLE SAFETY ANALYSES	The minimum water level in the SFP meets the assumptions of fuel handling or fuel cask drop accident analyses described in References 3 and 4 and are consistent with the assumptions of Regulatory Guide 1.25 (Ref. 5). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is well within the 10 CFR 100 (Ref. 6) limits. Reference 5 assumes there is 23 ft of water between the top of the damaged fuel assembly and the fuel pool surface for a fuel handling or fuel cask drop accident. This LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single assembly, dropped and lying horizontally on top of the spent fuel racks, there may be < 23 ft of water above the top of the assembly and the surface, by the width of the assembly. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rods fail from a hypothetical maximum drop. The SFP water level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

BASES

LCO

The specified water level preserves the assumptions of the fuel handling or fuel cask drop accident analyses. As such, it is the minimum required for movement of fuel assemblies or movement of a fuel cask in or over the SFP.

The LCO is modified by a Note which allows SFP level to be below the 647 ft elevation to support movement of a fuel cask in or over the SFP. This is necessary due to the water displaced by the fuel cask as it is lowered or dropped into the SFP. If the SFP level is normal prior to the fuel cask entering the SFP, the SFP could overflow.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the SFP or movement of a fuel cask in or over the SFP since the potential for a release of fission products exists.

ACTIONS

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

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

A.1 and A.2

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 or movement of a fuel cask in or over the SFP are immediately suspended. This effectively precludes a spent fuel handling or fuel cask drop accident from occurring. This does not preclude moving a fuel assembly or fuel cask to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient SFP water is available in the event of a fuel handling or fuel cask drop accident. The water level in the SFP must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable, based on operating experience.

During refueling operations, the level in the SFP is at equilibrium with that of the refueling cavity, and the level in the refueling cavity is checked daily in accordance with LCO 3.9.6, "Refueling Cavity Water Level."

REFERENCES

1. FSAR, Section 9.11
2. FSAR, Section 9.4
3. FSAR, Section 14.19
4. FSAR, Section 14.11
5. Regulatory Guide 1.25
6. 10 CFR 100.11

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND	As described in LCO 3.7.16, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 1720 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron.
APPLICABLE SAFETY ANALYSES	A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.16 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the SFP satisfies Criterion 2 of 10 CFR 50.36(c)(2).
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 SFP.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool until a complete spent fuel pool verification of the stored assemblies has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

BASES

ACTIONS

The 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 sufficient reason to require a reactor shutdown.

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

When the concentration of boron in the spent fuel pool 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 limit. Alternately, beginning a verification of the SFP fuel locations to ensure proper locations of the fuel can be performed.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR 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 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND	The spent fuel storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or used (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool is sized to store 892 irradiated fuel assemblies, which includes storage for failed fuel canisters. The spent fuel storage racks are grouped into two regions, Region I and Region II per Figure 3.7.16-1. The racks are designed as a Seismic Category I structure able to withstand seismic events. Region I contains racks in the spent fuel pool having a 10.25 inch center-to-center spacing and a single rack in the north tilt pit having an 11.25 inch by 10.69 inch center-to-center spacing. Region II contains racks in both the spent fuel pool and the north tilt pit having a 9.17 inch center-to-center spacing. Because of the smaller spacing and poison concentration, Region II racks have more limitations for fuel storage than Region I racks. Further information on these limitations can be found in Section 4.0, "Design Features." These limitations (e.g., enrichment, burnup) are sufficient to maintain a k_{eff} of $\leq$ 0.95 for spent fuel of original enrichment of up to 4.40%.
APPLICABLE SAFETY ANALYSES	The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Table 3.7.16-1, in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Table 3.7.16-1, in the accompanying LCO. Fuel assemblies not meeting the criteria of Table 3.7.16-1 shall be stored in accordance with Specification 4.3.1.1.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in Region II of either the spent fuel pool or the north tilt pit.

BASES

ACTIONS	Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 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, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown. <u>A.1</u> When the configuration of fuel assemblies stored in Region II the spent fuel pool is not in accordance with Table 3.7.16-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Table 3.7.16-1.
SURVEILLANCE REQUIREMENTS	<u>SR 3.7.16.1</u> This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Table 3.7.16-1 in the accompanying LCO prior to placing the fuel assembly in a Region II storage location.
REFERENCES	None

BASES

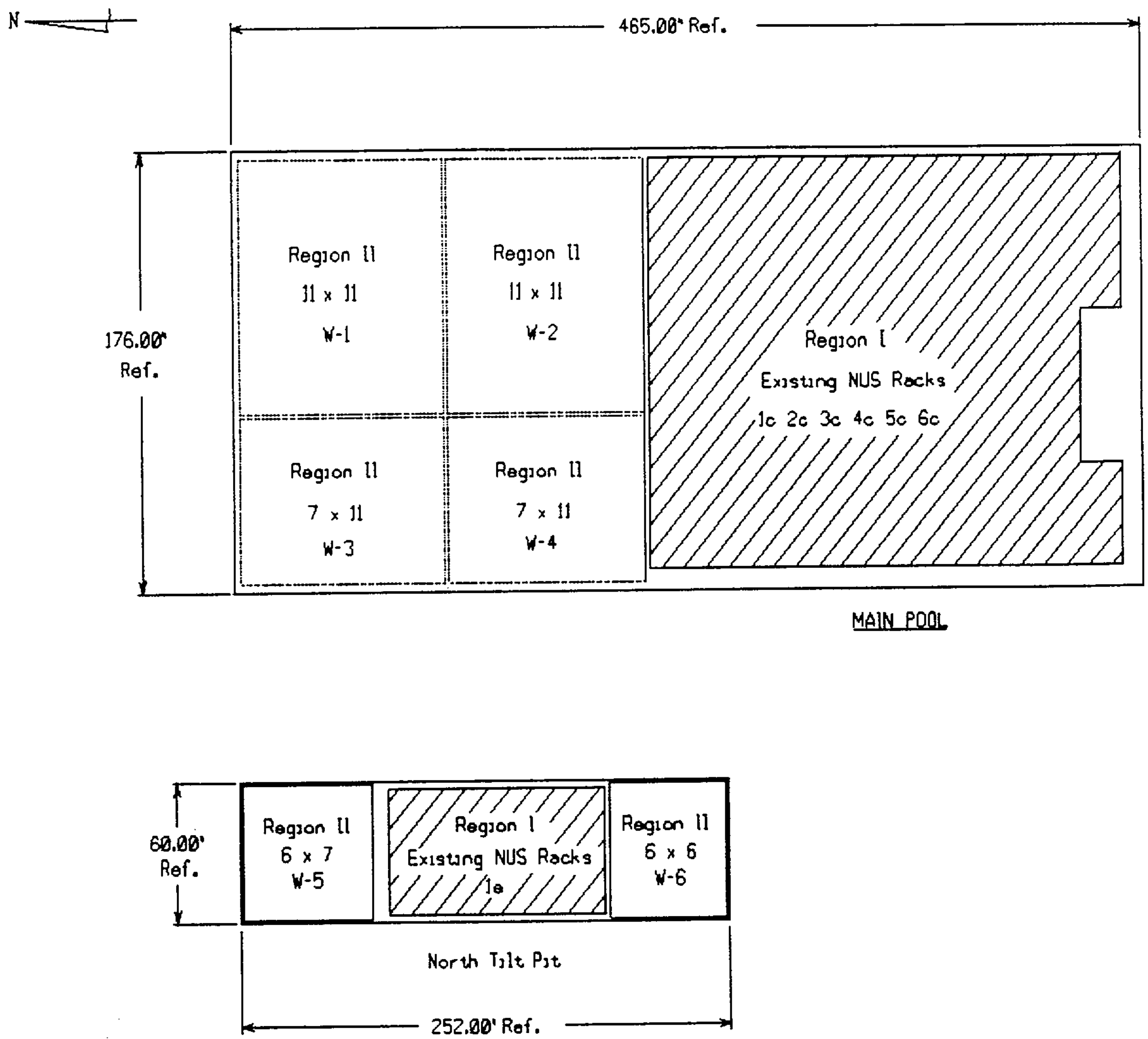

Figure B 3.7.16-1 (page 1 of 1)
Spent Fuel Arrangement

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

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

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

This limit is lower than the activity value that might be expected from a 1 gpm tube leak of primary coolant at the limit of 1.0 µCi/gm as assumed in the safety analyses with exception of the control rod ejection analysis which assumes 0.6 gpm. LCO 3.4.13, "PCS Operational LEAKAGE," is more restrictive in that the limit for a primary to secondary tube leak is 0.3 gpm. The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and primary coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

Operating a plant at the allowable limits would result in a 2 hour Exclusion Area Boundary (EAB) exposure well within the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the Main Steam Line Break (MSLB), outside of containment as discussed in the FSAR, Chapter 14.14 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB are well within the plant EAB limits (Ref. 1) for whole body and thyroid dose rates.

BASES

APPLICABLE SAFETY ANALYSES (continued)	With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the primary coolant temperature and pressure have decreased sufficiently for the Shutdown Cooling System to complete the cooldown. In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through MSSVs and ADVs during the event. Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).
LCO	As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 limits the radiological consequences of a Design Basis Accident (DBA) to well within the required limit (Ref. 1). Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the plant 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 PCS and steam generators are at low pressure or depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant is an indication of a problem in the PCS and contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits in the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

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 primary coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11
2. FSAR, Section 14.14

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The plant Class 1E Electrical Power Distribution System AC sources consist of the offsite power sources, and the onsite standby power sources, Diesel Generators 1-1 and 1-2 (DGs). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The AC power system at Palisades consists of a 345 kV switchyard, three circuits connecting the plant with off-site power (station power, startup, and safeguards transformers), the on-site distribution system, and two DGs. The on-site distribution system is divided into safety related (Class 1-E) and non-safety related portions.

The switchyard interconnects six transmission lines from the off-site transmission system and the output line from the Palisades main generator. These lines are connected in a "breaker and a half" scheme between the Front (F) and Rear (R) buses such that any single off-site line may supply the Palisades station loads when the plant is shutdown.

Two circuits supplying Palisades 2400 V buses from off-site are fed directly from a switchyard bus through the startup and safeguards transformers. They are available both during operation and during shutdown. The third circuit supplies the plant loads by "back feeding" through the main generator output circuit and station power transformers after the generator has been disconnected by a motor operated disconnect.

The station power transformers are connected into the main generator output circuit. Station power transformers 1-1 and 1-2 connect to the generator 22 kV output bus. Station power transformer 1-3 connects to the generator output line on the high voltage side of the main transformer. Station power transformers 1-1 and 1-3 supply non-safety related 4160 V loads during plant power operation and during backfeeding operations. Station power transformer 1-2 can supply both safety related and non-safety related 2400 V loads during plant power operation or backfeeding operation.

BACKGROUND (continued)

The three startup transformers are connected to a common 345 kV overhead line from the switchyard R bus. Startup transformers 1-1 and 1-3 supply 4160 V non-safety related station loads; Startup Transformer 1-2 can supply both safety related and non-safety related 2400 V loads. The startup transformers are available during operation and shutdown.

Safeguards Transformer 1-1 is connected to the switchyard F bus. It feeds station 2400 V loads through an underground line. It is available to supply these loads during operation and shutdown.

The onsite distribution system consists of seven main distribution buses (4160 V buses 1A, 1B, 1F, and 1G, and 2400 V buses 1C, 1D, and 1E) and supported lower voltage buses, Motor Control Centers (MCCs), and lighting panels. The 4160 V buses and 2400 V bus 1E are not safety related. Buses 1C and 1D and their supported buses and MCCs form two independent, redundant, safety related distribution trains. Each distribution train supplies one train of engineered safety features equipment.

In the event of a generator trip, all loads supplied by the station power transformers are automatically transferred to the startup transformers. Loads supplied by the safeguards transformer are unaffected by a plant trip. If power is lost to the safeguards transformer, the 2400 V loads will automatically transfer to startup transformer 1-2. If the startup transformers are not energized when these transfers occur, their output breakers will be blocked from closing and the 2400 V safety related buses will be energized by the DGs.

The two DGs each supply one 2400 V bus. They provide backup power in the event of loss of off-site power, or loss of power to the associated 2400 V bus. The continuous rating of the DGs is 2500 kW, with 110 percent overload permissible for 2 hours in any 24 hour period. The required fuel in the Fuel Oil Storage Tank and DG Day Tank will supply one DG for a minimum period of 7 days assuming accident loading conditions and fuel conservation practices.

If either 2400 V bus, 1C or 1D, experiences a sustained undervoltage, the associated DG is started, the affected bus is separated from its offsite power sources, major loads are stripped from that bus and its supported buses, the DGs are connected to the bus, and ECCS or shutdown loads are started by an automatic load sequencer.

BACKGROUND (continued)	The DGs share a common fuel oil storage and transfer system. A single buried Fuel Oil Storage Tank is used, along with an individual day tank for each DG, to maintain the required fuel oil inventory. Two fuel transfer pumps are provided. The fuel transfer pumps are necessary for long term operation of the DGs. Testing has shown that each DG consumes about 2.6 gallons of fuel oil per minute at 2400 kW. Each day tank is required to contain at least 2500 gallons. Therefore, each fuel oil day tank contains sufficient fuel for more than 15 hours of full load (2500 kW) operation. Beyond that time, a fuel transfer pump is required for continued DG operation.

Either fuel transfer pump is capable of supplying either DG. However, each fuel transfer pump is not capable, with normally available switching, of being powered from either DG. DG 1-1 can power either fuel transfer pump, but DG 1-2 can only power P-18A. The fuel oil pumps share a common fuel oil storage tank, and common piping.

Fuel transfer pump P-18A is powered from MCC-8, which is normally connected to Bus 1D (DG 1-2) through Station Power Transformer 12 and Load Center 12. In an emergency, P-18A can be powered from Bus 1C (DG 1-1) by cross connecting Load Centers 11 and 12.

Fuel transfer pump P-18B is powered from MCC-1, which is normally connected to Bus 1C (DG 1-1) through Station Power Transformer 19 and Load Center 19. P-18B cannot be powered, using installed equipment, from Bus 1D (DG 1-2).

APPLICABLE SAFETY ANALYSES

The safety analyses do not explicitly address AC electrical power. They do, however, assume that the Engineered Safety Features (ESF) are available. The OPERABILITY of the ESF functions is supported by the AC Power Sources.

The design requirements are for each assumed safety function to be available under the following conditions:

a. The occurrence of an accident or transient,

b. The resultant consequential failures,

c. A worst case single active failure,

d. Loss of all offsite or all onsite AC power, and

e. The most reactive control rod fails to insert.

BASES

APPLICABLE SAFETY ANALYSES (continued)

One proposed mechanism for the loss of off-site power is a perturbation of the transmission grid because of the loss of the plant's generating capacity. A loss of off-site power as a result of a generator trip can only occur during MODE 1 with the generator connected to the grid. However, it is also assumed in analysis for some events in MODE 2, such as a control rod ejection. No specific mechanism for initiating a loss of off-site power when the plant is not on the line is discussed in the FSAR.

In most cases, it is conservative to assume that off-site power is lost concurrent with the accident and that the single failure is that of a DG. That would leave only one train of safeguards equipment to cope with the accident, the other being disabled by the loss of AC power. Those analyses which assume that a loss of off-site power and failure of a single DG accompany the accident assume 11 seconds from the loss of power until the bus is re-energized. This time includes time for all portions of the circuitry necessary for detecting the undervoltage (relays and auxiliary relays) and starting the DG. Included in the 11 seconds, the analyses also assume 10 seconds for the DG to start and connect to the bus, and additional time for the sequencer to start each safeguards load.

The same assumptions are not conservative for all accident analyses. When analyzing the effects of a steam or feed line break, the loss of the condensate and feedwater pumps would reduce the steam generator inventory, so a loss of off-site power is not assumed.

In MODE 5 and MODE 6, loss of off-site power can be considered as an initiating event for a loss of shutdown cooling event.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System and an independent DG for each safeguards train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence or a postulated DBA.

BASES

LCO (continued)

General Design Criterion 17 (Ref. 1) requires, in part, that: "Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions."

The qualified offsite circuits available are Safeguards Transformer 1-1 and Startup Transformer 1-2. Station Power Transformer 1-2 is not qualified as a required source for LCO 3.8.1 since it is not independent of the other two offsite circuits. This LCO does not prohibit use of Station Power Transformer to power the 2400 V safety related buses, but the two qualified sources must be OPERABLE.

Each offsite circuit must be capable of maintaining acceptable frequency and voltage, and accepting required loads during an accident, while supplying the 2400 V safety related buses.

Following a loss of offsite power, each DG must be capable of starting and connecting to its respective 2400 V bus. This will be accomplished within 10 seconds after receipt of a DG start signal. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the 2400 V safety related buses.

Proper sequencing of loads and tripping of nonessential loads are required functions for DG OPERABILITY.

APPLICABILITY

The AC sources are required to be OPERABLE above MODE 5 to ensure that redundant sources of off-site and on-site AC power are available to support engineered safeguards equipment in the event of an accident or transient. The AC sources also support the equipment necessary for power operation, plant heatups and cooldowns, and shutdown operation.

The AC source requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.2, "AC Sources - Shutdown."

BASES

ACTIONS

A.1

To ensure a highly reliable power source remains with the one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in failure to meet this Required Action. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.8.1 A.1 must be initially performed within 1 hour without any SR 3.0.2 extension, subsequent performances at the "Once per 8 hours" interval may utilize the 25% SR 3.0.2 extension.

A.2

According to the recommendations of Regulatory Guide (RG) 1.93 (Ref. 2), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the plant safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

BASES

ACTIONS

A.2 (continued)

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. The second Completion Time for Required Action A.2 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable, and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 7 days (for a total of 17 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 10 day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

The Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

BASES

ACTIONS (continued)

B.2

In accordance with LCO 3.0.6, the requirement to declare required features inoperable carries with it the requirement to take those actions required by the LCO for that required equipment.

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features within a train redundant to the train that has an inoperable DG. If the train that has an inoperable DG contains multiple features redundant to the inoperable feature in the other train, all those multiple features must be declared inoperable. For example, if DG 1-1 and Containment Spray Pump P-54A are inoperable concurrently, Containment Spray Pumps P-54B and P-54C must both be declared inoperable. In this example, if off-site power were lost, neither P-54B nor P-54C would be available.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required supporting or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for Required Action B.2. Four hours from the discovery of these events existing concurrently, is acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown.

In this Condition, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost.

BASES

ACTIONS

B.2 (continued)

The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 (test starting of the OPERABLE DG) does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed to not exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing Required Action B.3.1 or B.3.2 the corrective action system would normally continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B. According to Generic Letter 84-15 (Ref. 3), 24 hours is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

BASES

ACTIONS (continued)

B.4

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System for a limited period. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently returned OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 7 day and 10 day Completion Time means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

BASES

ACTIONS (continued)

C.1

In accordance with LCO 3.0.6 the requirement to declare required features inoperable carries with it the requirement to take those actions required by the LCO for that required equipment.

Required Action C.1, which applies when two required offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours. The rationale for the reduction to 12 hours is that RG 1.93 (Ref. 2) recommends a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable), a required feature becomes inoperable, this Completion Time begins to be tracked.

BASES

ACTIONS (continued)

C.2

According to the recommendations of RG 1.93 (Ref. 2), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to accomplish a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the plant in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition D to provide the requirements for the loss of one offsite circuit and one DG without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

According to the recommendations of RG 1.93 (Ref. 2), operation may continue in Condition D for a period that should not exceed 12 hours.

BASES

ACTIONS (continued)

E.1

With both DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, no AC source would be available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since an inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to the recommendations of RG 1.93 (Ref. 2), with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

If the inoperable AC power sources cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

G.1

Condition G corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

BASES

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 4). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of RG 1.9 (Ref. 5) and RG 1.137 (Ref. 6).

Where the SRs discussed herein specify voltage and frequency tolerances for the DGs operated in the "Unit" mode, the following is applicable. The minimum steady state output voltage of 2280 V is 95% of the nominal 2400 V generator rating. This value is above the setting of the primary undervoltage relays (127-1 and 127-2) and above the minimum analyzed acceptable bus voltage. It also allows for voltage drops to motors and other equipment down through the 120 V level. The specified maximum steady state output voltage of 2520 V is 105% of the nominal generator rating of 2400 V. It is below the maximum voltage rating of the safeguards motors, 2530 V. The specified minimum and maximum frequencies of the DG are 59.5 Hz and 61.2 Hz, respectively. The minimum value assures that ESF pumps provide sufficient flow to meet the accident analyses. The maximum value is equal to 102% of the 60 Hz nominal frequency and is derived from the recommendations given in RG 1.9 (Ref. 5).

Higher maximum tolerances are specified for final steady state voltage and frequency following a loss of load test, because that test must be performed with the DG controls in the "Parallel" mode. Since "Parallel" mode operation introduces both voltage and speed droop, the DG final conditions will not return to the nominal "Unit" mode settings.

SR 3.8.1.1

This SR assures that the required offsite circuits are OPERABLE. Each offsite circuit must be energized from associated switchyard bus through its disconnect switch to be OPERABLE.

Since each required offsite circuit transformer has only one possible source of power, the associated switchyard bus, and since loss of voltage to either the switchyard bus or the transformer is alarmed in the control room, correct alignment and voltage may be verified by the absence of these alarms.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.1 (continued)

The 7 day Frequency is adequate because disconnect switch positions cannot change without operator action and because their status is displayed in the control room.

SR 3.8.1.2

This SR helps to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition.

The monthly test starting of the DG provides assurance that the DG would start and be ready for loading in the time period assumed in the safety analyses. The monthly test, however does not, and is not intended to, test all portions of the circuitry necessary for automatic starting and loading. The operation of the bus undervoltage relays and their auxiliary relays which initiate DG starting, the control relay which initiates DG breaker closure, and the DG breaker closure itself are not verified by this test. Verification of automatic operation of these components requires de-energizing the associated 2400 V bus and cannot be done during plant operation. For this test, the 10 second timing is started when the DG receives a start signal, and ends when the DG voltage sensing relays actuate. For the purposes of SR 3.8.1.2, the DGs are manually started from standby conditions. Standby conditions for a DG mean the diesel engine is not running, its coolant and oil temperatures are being maintained consistent with manufacturer recommendations, and ≥ 20 minutes have elapsed since the last DG air roll.

Three relays sense the terminal voltage on each DG. These relays, in conjunction with a load shedding relay actuated by bus undervoltage, initiate automatic closing of the DG breaker. During monthly testing, the actuation of the three voltage sensing relays is used as the timing point to determine when the DG is ready for loading.

The 31 day Frequency for performance of SR 3.8.1.2 agrees with the original licensing basis for the Palisades plant.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads for at least 15 minutes. A minimum total run time of 60 minutes is required to stabilize engine temperatures.

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because the load shedding circuits, which are actuated by the 2400 V bus undervoltage relays and which must function to initiate automatic DG loading, are blocked when the DG breaker is closed. This load shed block assures that a spurious undervoltage will not cause load shedding while a DG is the sole source for accident loads, but it prevents automatic DG actuation while the DG is paralleled to the grid.

The 31 day Frequency for this Surveillance is consistent with the original Palisades licensing basis.

The SR is modified by three Notes. Note 1 states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated. Note 2 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 3 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The specified level is adequate for a minimum of 15 hours of DG operation at full load.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and plant operators would be aware of any uses of the DG during this period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.5

Each DG is provided with an engine overspeed trip to prevent damage to the engine. The loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. This Surveillance may be accomplished with the DG in the "Parallel" mode.

An acceptable method is to parallel the DG with the grid and load the DG to a load equal to or greater than its single largest post-accident load. The DG breaker is tripped while its voltage and frequency (or speed) are being recorded. The time, voltage, and frequency tolerances specified in this SR are derived from the recommendations of RG 1.9, Revision 3 (Ref. 5).

RG 1.9 recommends that the increase in diesel speed during the transient does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower. The Palisades DGs have a synchronous speed of 900 rpm and an overspeed trip setting range of 1060 to 1105 rpm. Therefore, the maximum acceptable transient frequency for this SR is 68 Hz.

The minimum steady state voltage is specified to provide adequate margin for the switchgear and for both the 2400 and 480 V safeguards motors; the maximum steady state voltage is 2400 +10% V as recommended by RG 1.9.

The minimum acceptable frequency is specified to assure that the safeguards pumps powered from the DG would supply adequate flow to meet the safety analyses. The maximum acceptable steady state frequency is slightly higher than the +2% (61.2 Hz) recommended by RG 1.9 because the test must be performed with the DG controls in the Parallel mode. The increased frequency allowance of 0.3 Hz is based on the expected speed differential associated with performance of the test while in the "Parallel" mode.

The 18 month surveillance Frequency is consistent with the recommendation of RG 1.9 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine and generator load response under a complete loss of load. These acceptance criteria provide DG damage protection. The 4000 V limitation is based on generator rating of 2400/4160V and the ratings of those components (connecting cables and switchgear) which would experience the voltage transient. While the DG is not expected to experience this transient during an event and continue to be available, this response ensures that the DG is not degraded for future application, including re-connection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, yet still provide adequate testing margin between the specified power factor limit and the DG design power factor limit of 0.8, testing must be performed using a power factor $\leq$ 0.9. This is consistent with RG 1.9 (Ref. 5).

The 18 month Frequency is consistent with the recommendation of RG 1.9 and is intended to be consistent with expected fuel cycle lengths.

SR 3.8.1.7

As recommended by RG 1.9 (Ref. 5) this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and re-energizing of the emergency buses and respective loads from the DG.

The requirement to energize permanently connected loads is met when the DG breaker closes, energizing its associated 2400 V bus. Permanently connected loads are those which are not disconnected from the bus by load shedding relays. They are energized when the DG breaker closes. It is not necessary to monitor each permanently connected load.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.7 (continued)

The DG auto-start and breaker closure time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. For this test, the 10 second timing is started when the DG receives a start signal, and ends when the DG breaker closes. The safety analyses assume 11 seconds from the loss of power until the bus is re-energized.

The requirement to verify that auto-connected shutdown loads are energized refers to those loads which are actuated by the Normal Shutdown Sequencer. Each load should be started to assure that the DG is capable of accelerating these loads at the intervals programmed for the Normal Shutdown Sequence. The sequenced pumps may be operating on recirculation flow.

The requirements to maintain steady state voltage and frequency apply to the "steady state" period after all sequenced loads have been started. This period need only be long enough to achieve and measure steady voltage and frequency.

The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved. The requirement to supply permanently connected loads for ≥ 5 minutes, refers to the duration of the DG connection to the associated safeguards bus. It is not intended to require that sequenced loads be operated throughout the 5 minute period. It is not necessary to monitor each permanently connected load.

The requirement to verify the connection and supply of permanently and automatically connected loads is intended to demonstrate the DG loading logic. This testing may be accomplished in any series of sequential, overlapping, or total steps so that the required connection and loading sequence is verified.

The Frequency of 18 months is consistent with the recommendations of RG 1.9 (Ref. 5).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8

RG 1.9 (Ref. 5) recommends demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 100 minutes of which is at a load above its analyzed peak accident loading and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The 100 minutes required by the SR satisfies the intent of the recommendations of the RG, but allows some tolerance between the time requirement and the DG rating. Without this tolerance, the load would have to be reduced at precisely 2 hours to satisfy the SR without exceeding the manufacturer's rating of the DG.

The DG starts for this Surveillance can be performed either from standby or hot conditions.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, yet still provide adequate testing margin between the specified power factor limit and the DG design power factor limit of 0.8, testing must be performed using a power factor of ≤ 0.9. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

In addition, a Note to the SR states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated.

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because the load shedding circuits, which are actuated by the 2400 V bus undervoltage relays and which must function to initiate automatic DG loading, are blocked when the DG breaker is closed. This load shed block assures that a spurious undervoltage will not cause load shedding while a DG is the sole source for accident loads, but it prevents automatic DG actuation while the DG is paralleled to the grid.

The 18 month Frequency is consistent with the recommendations of RG 1.9 (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

As recommended by RG 1.9 (Ref. 5), this Surveillance ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and that the DG can be returned to ready to load status when offsite power is restored. The test is performed while the DG is supplying its associated 2400 V bus, but not necessarily carrying the sequenced accident loads. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open, the automatic load sequencer is reset, and the DG controls are returned to "Unit."

During the period when the DG is paralleled to the grid, it must be considered inoperable. This is because the load shedding circuits, which are actuated by the 2400 V bus undervoltage relays and which must function to initiate automatic DG loading, are blocked when the DG breaker is closed. This load shed block assures that a spurious undervoltage will not cause load shedding while a DG is the sole source for accident loads, but it prevents automatic DG actuation while the DG is paralleled to the grid.

The Frequency of 18 months is consistent with the recommendations of RG 1.9 (Ref. 3).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.10

If power is lost to bus 1C or 1D, loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs by concurrent motor starting currents. The 0.3 second load sequence time tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and ensures that safety analysis assumptions regarding ESF equipment time delays are met. Logic Drawing E-17 Sheet 4 (Ref. 7) provides a summary of the automatic loading of safety related buses.

The Frequency of 18 months is consistent with the recommendations of RG 1.9 (Ref. 5), takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.11

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, PCS, and containment design limits are not exceeded.

The requirement to energize permanently connected loads is met when the DG breaker closes, energizing its associated 2400 V bus. Permanently connected loads are those which are not disconnected from the bus by load shedding relays. They are energized when the DG breaker closes. It is not necessary to monitor each permanently connected load. The DG auto-start and breaker closure time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. For this test, the 10 second timing is started when the DG receives a start signal, and ends when the DG breaker closes. The safety analyses assume 11 seconds from the loss of power until the bus is re-energized.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

In addition, a Note to the SR states that momentary transients outside the required band do not invalidate this test. This is to assure that a minor change in grid conditions and the resultant change in DG load, or a similar event, does not result in a surveillance being unnecessarily repeated.

The requirement to verify that auto-connected shutdown loads are energized refers to those loads which are actuated by the DBA Sequencer. Each load should be started to assure that the DG is capable of accelerating these loads at the intervals programmed for the DBA Sequence. Since the containment spray pumps do not actuate on SIS generated by Pressure Low Pressure, the test should be performed such that spray pump starting by the sequencer is also verified along with the other SIS loads. The sequenced pumps may be operating on recirculation flow or in other testing modes. The requirements to maintain steady state voltage and frequency apply to the "steady state" period after all sequenced loads have been started. This period need only be long enough to achieve and measure steady voltage and frequency.

The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved. The requirement to supply permanently connected loads for ≥ 5 minutes, refers to the duration of the DG connection to the associated 2400 V bus. It is not intended to require that sequenced loads be operated throughout the 5 minute period. It is not necessary to monitor each permanently connected load.

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17
2. Regulatory Guide 1.93, December 1974
3. Generic Letter 84-15, July 2, 1984
4. 10 CFR 50, Appendix A, GDC 18
5. Regulatory Guide 1.9, Rev. 3, July 1993
6. Regulatory Guide 1.137, Rev. 1, October 1979
7. Palisades Logic Drawing E-17, Sheet 4

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND	A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."
APPLICABLE SAFETY ANALYSES	The safety analyses do not explicitly address electrical power. They do, however, assume that various electrically powered and controlled equipment is available. Electrical power is necessary to terminate and mitigate the effects of many postulated events which could occur in MODES 5 and 6. Analyzed events which might occur during MODES 5 and 6 are Loss of PCS inventory or Loss of PCS Flow, (which in MODES 5 and 6 would be grouped as a Loss of Shutdown Cooling event), and radioactive releases (Fuel Handling Accident, Cask Drop, Radioactive Gas Release, Etc.). In general, when the plant is shut down, the Technical Specifications requirements ensure that the plant has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed above MODE 5 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the primary coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced, and in minimal consequences. The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	This LCO requires one offsite circuit to be OPERABLE. One OPERABLE offsite circuit ensures that all required loads may be powered from offsite power. Since only one offsite AC source is required, independence is not a criterion. Any of the three offsite supplies, Safeguards Transformer 1-1, Station Power Transformer 1-2, or Startup Transformer 1-2 is acceptable as a qualified circuit.

BASES

LCO (continued)

An OPERABLE DG, associated with a distribution subsystem required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit.

Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling).

The DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective 2400 V bus on detection of bus undervoltage, and accepting required loads. Proper "Normal Shutdown" loading sequence, and tripping of nonessential loads, is a required function for DG OPERABILITY. A Service Water Pump must be started soon after the DG to assure continued DG operability. The DBA loading sequence is not required to be OPERABLE since the Safety Injection Signal is disabled during MODES 5 and 6.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

d. Monitor and maintain the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODES 5 and 6. This LCO provides the necessary ACTIONS if the AC electrical power sources required by this LCO become unavailable during movement of irradiated fuel assemblies.

The AC source requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.1, "AC Sources - Operating."

BASES

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to supply the 2400 V safety related bus or buses required by LCO 3.8.10. Since the required offsite AC source is only required to support features required by other LCOs, the option to declare those required features with no offsite power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Action A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of AC sources.

With the required DG inoperable, the minimum required diversity of AC power sources is not available.

Required Actions A.2.1 through A.2.4, and B.1 through B.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources (and to continue this action until restoration is accomplished) in order to provide the necessary AC power to the plant safety systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required AC power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6.

The SRs from LCO 3.8.1 which are required are those which both support a feature required in MODES 5 and 6 and which can be performed without affecting the OPERABILITY or reliability of the required sources.

With only one DG available, many tests cannot be performed since their performance would render that DG inoperable during the test. This is the case for tests which require DG loading: SRs 3.8.1.3, 3.8.1.5, 3.8.1.6, 3.8.1.7, 3.8.1.8, 3.8.1.9, 3.8.1.10, and 3.8.1.11.

REFERENCES

None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel, Lube Oil, and Starting Air

BASES

BACKGROUND

The Diesel Generators (DGs) are provided with a storage tank having a required fuel oil inventory sufficient to operate one diesel for a period of 7 days, while the DG is supplying maximum post-accident loads. This onsite fuel oil capacity is sufficient to operate the DG for longer than the time to replenish the onsite supply from offsite sources.

Fuel oil is transferred from the Fuel Oil Storage Tank to either day tank by either of two Fuel Transfer Systems. The fuel oil transfer system which includes fuel transfer pump P-18A can be powered by offsite power, or by either DG. However, the fuel oil transfer system which includes fuel transfer pump P-18B can only be powered by offsite power, or by DG 1-1.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide (RG) 1.137 (Ref. 1) addresses the recommended fuel oil practices as supplemented by ANSI N195-1976 (Ref. 2).

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation. This supply is sufficient supply to allow the operator to replenish lube oil from offsite sources. Implicit in this LCO is the requirement to assure, though not necessarily by testing, the capability to transfer the lube oil from its storage location to the DG oil sump, while the DG is running.

Each DG is provided with an associated starting air subsystem to assure independent start capability. The starting air system is required to have a minimum capacity with margin for a DG start attempt without recharging the air start receivers.

BASES

APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating"; during MODES 5 and 6, in the Bases for LCO 3.8.2, "AC Sources - Shutdown." Since diesel fuel, lube oil, and starting air subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	Stored diesel fuel oil is required to have sufficient supply for 7 days of full accident load operation. It is also required to meet specific standards for quality. The specified 7 day requirement and the 6 day quantity listed in Condition A are taken from the Engineering Analysis associated with Event Report E-PAL-93-026B. Additionally, the ability to transfer fuel oil from the storage tank to each day tank is required from each of the two transfer pumps. Additionally, sufficient lube oil supply must be available to ensure the capability to operate at full accident load for 7 days. This requirement is in addition to the lube oil contained in the engine sump. The specified 7 day requirement and the 6 day quantity listed in Condition B are based on an assumed lube oil consumption of 0.8 to 1.0% of fuel oil consumption. The starting air subsystem must provide, without the aid of the refill compressor, sufficient air start capacity, including margin, to assure start capability for its associated DG. These requirements, in conjunction with an ability to obtain replacement supplies within 7 days, support the availability of the DGs. DG day tank fuel requirements are addressed in LCOs 3.8.1 and 3.8.2.
APPLICABILITY	DG OPERABILITY is required by LCOs 3.8.1 and 3.8.2 to ensure the availability of the required AC power to shut down the reactor and maintain it in a safe shutdown condition following a loss of off-site power. Since diesel fuel, lube oil, and starting air support LCOs 3.8.1 and 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits, and the fuel transfer system is required to be OPERABLE, when either DG is required to be OPERABLE.

BASES

ACTIONS

A.1

In this Condition, the available DG fuel oil supply is less than the required 7 day supply, but enough for at least 6 days. This condition allows sufficient time to obtain additional fuel and to perform the sampling and analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required inventory prior to declaring the DGs inoperable.

B.1

In this Condition, the available DG lube oil supply in storage is less than the required 7 day supply, but enough for at least 6 days. This condition allows sufficient time to obtain additional lube oil. A period of 48 hours is considered sufficient to complete restoration of the required inventory prior to declaring the DGs inoperable.

C.1, D.1, and E.1

Since DG 1-2 cannot power fuel transfer pump P-18B, without P-18A, DG 1-2 becomes dependant on offsite power or DG 1-1 for its fuel supply (beyond the 15 hours it will operate on the day tank), and does not meet the requirement for independence. Since the condition is not as severe as the DG itself being inoperable, 15 hours is allowed to restore the fuel transfer system to operable status prior to declaring the DG inoperable.

Without P-18B, either DG can still provide power to the remaining fuel transfer system. Therefore, neither DG is directly affected. Continued operation with a single remaining fuel transfer system, however, must be limited since an additional single active failure (P-18A) could disable the onsite power system.. Because the loss of P-18B is less severe than the loss of one DG, a 7 day Completion Time is allowed.

If both fuel transfer systems are inoperable, the onsite AC sources are limited to about 15 hours duration. Since this condition is not as severe as both DGs being inoperable, 8 hours is allowed to restore one fuel transfer pump to OPERABLE status.

BASES

ACTIONS (continued)

F.1

With the stored fuel oil properties, other than viscosity, and water and sediment, defined in the Fuel Oil Testing Program not within the required limits, but acceptable for short term DG operation, a period of 30 days is allowed for restoring the stored fuel oil properties. The most likely cause of stored fuel oil becoming out of limits is the addition of new fuel oil with properties that do not meet all of the limits. This 30 day period provides sufficient time to determine if new fuel oil, when mixed with stored fuel oil, will produce an acceptable mixture, or if other methods to restore the stored fuel oil properties are required. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

G.1

With a Required Action and associated Completion Time not met, or with diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than addressed by Conditions A, B, or F, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

In the event that diesel fuel oil with viscosity, or water and sediment is out of limits, this would be unacceptable for even short term DG operation. Viscosity is important primarily because of its effect on the handling of the fuel by the pump and injector system; water and sediment provides an indication of fuel contamination. When the fuel oil stored in the Fuel Oil Storage Tank is determined to be out of viscosity, or water and sediment limits, the DGs must be declared inoperable, immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tank to support either DG's operation for 7 days at full post-accident load. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 24 hour Frequency is specified to ensure that a sufficient supply of fuel oil is available, since the Fuel Oil Storage Tank is the fuel oil supply for the diesel fire pumps, heating boilers, and rad waste evaporators, in addition to the DGs.

SR 3.8.3.2

This Surveillance ensures that sufficient stored lube oil inventory is available to support at least 7 days of full accident load operation for one DG. The 200 gallons requirement is based on an estimated consumption of 0.8 to 1.0% of fuel oil consumption.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run times are closely monitored by the plant staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil and stored fuel oil are of the appropriate grade and have not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion.

Testing for viscosity, specific gravity, and water and sediment is completed for fuel oil delivered to the plant prior to its being added to the Fuel Oil Storage Tank. Fuel oil which fails the test, but has not been added to the Fuel Oil Storage Tank does not imply failure of this SR and requires no specific action. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tank without concern for contaminating the entire volume of fuel oil in the storage tank.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

Fuel oil is tested for other of the parameters specified in ASTM D975 (Ref. 3) in accordance with the Fuel Oil Testing Program required by Specification 5.5.11. Fuel oil determined to have one or more measured parameters, other than viscosity or water and sediment, outside acceptable limits will be evaluated for its effect on DG operation. Fuel oil which is determined to be acceptable for short term DG operation, but outside limits will be restored to within limits in accordance with LCO 3.8.3 Condition F.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The pressure specified in this SR is intended to reflect the acceptable margin from which successful starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the Fuel Oil Storage Tank once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it reduces the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies and acceptance criteria are established in the Fuel Oil Testing Program based, in part, on those recommended by RG 1.137 (Ref. 1). This SR is for preventative maintenance.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.5 (continued)

The presence of water does not necessarily represent failure of this SR provided the accumulated water is removed in accordance with the requirements of the Fuel Oil Testing Program.

SR 3.8.3.6

This SR demonstrates that each fuel transfer pump and the fuel transfer system controls operate and control transfer of fuel from the Fuel Oil Storage Tank to each day tank and engine mounted tank. This is required to support continuous operation of standby power sources.

This SR provides assurance that the following portions of the fuel transfer system is OPERABLE:

a. Fuel Transfer Pumps;

b. Day and engine mounted tank filling solenoid valves; and

c. Day and engine mounted tank automatic level controls.

The 92 day Frequency corresponds to the testing requirements for pumps in the ASME Code, Section XI (Ref. 4). Additional assurance of fuel transfer system OPERABILITY is provided during the monthly starting and loading tests for each DG when the fuel oil system will function to maintain level in the day and engine mounted tanks.

REFERENCES

1. Regulatory Guide 1.137

2. ANSI N195-1976, Appendix B

3. ASTM Standards, D975, Table 1

4. ASME, Boiler and Pressure Vessel Code, Section XI

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC power system with control power. It also provides control power to selected safety related equipment and power to the preferred AC Buses (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure.

The 125 V DC electrical power system consists of two independent and redundant safety related Class 1E DC power sources. Each DC source consists of one 125 V battery, one battery charger, and the associated control equipment and interconnecting cabling. While each station battery has two associated battery chargers, one powered by the associated AC power distribution system (the directly connected chargers), and one powered from the opposite AC power distribution system (the cross connected chargers), the cross connect chargers are not required to be OPERABLE and cannot be credited to meet this LCO. The battery chargers are normally operated in pairs, either both direct connected chargers or both cross connected chargers, to assure a diverse AC supply.

During normal operation, the 125 V DC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power from the battery charger, the DC load continues to be powered from the station batteries.

The DC power distribution system is described in the Bases for LCO 3.8.9, "Distributions System - Operating."

Each battery has adequate storage capacity to carry the required load continuously for at least 4 hours and to perform three complete cycles of intermittent loads discussed in the FSAR, Chapter 8 (Ref. 2).

Each 125 V battery is separately housed in a ventilated room apart from its charger and distribution centers. Each DC source is separated physically and electrically from the other DC source to ensure that a single failure in one source does not cause a failure in a redundant source.

BASES

BACKGROUND (continued)	The batteries for the DC power sources are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 125.7 V per battery discussed in the FSAR, Chapter 8 (Ref. 2). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 3). Each DC electrical power source has ample power output capacity for the steady state operation of connected loads during normal operation, while at the same time maintaining its battery fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 2).
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating." The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The DC power sources, each consisting of one battery, one directly connected battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of DC control power and Preferred AC power to shut down the reactor and maintain it in a safe condition. An OPERABLE DC electrical power source requires its battery to be OPERABLE and connected to the associated DC bus. In order for the battery to remain OPERABLE for any extended period of time, at least one charger must be in service. Without a charger in service, the DC loads would reduce the battery charge to the point where the battery would become inoperable. Disconnecting a charger, however, does not, in itself, make a battery inoperable. The LCO requires chargers ED-15 and ED-16 because those chargers are powered by the AC power distribution system and DG associated with the battery they supply. If only the cross connected chargers were available, and a loss of off-site power should occur concurrently with the loss of one DG, both safeguards trains would eventually become disabled. One train would be disabled by the lack of AC motive power;

BASES

LCO (continued)

the other would become disabled when the battery, whose only OPERABLE charger is fed by the failed DG, became depleted.

The required chargers, ED-15 and ED-16, must be OPERABLE, but need not actually be in service because the probability of a concurrent loss of offsite power with loss of one DG is low, and battery charging current is not needed immediately after an accident.

APPLICABILITY

The DC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that redundant sources of DC power are available to support engineered safeguards equipment and plant instrumentation in the event of an accident or transient. The DC sources also support the equipment and instrumentation necessary for power operation, plant heatups and cooldowns, and shutdown operation.

The DC source requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1 and A.2

With one of the required chargers (ED-15 or ED-16) inoperable, the cross-connected charger must be placed in service within 2 hours, if it is not already in service, to maintain the battery in OPERABLE status. If the cross-connect charger is not placed in service within 2 hours, Condition C would be entered.

Additionally for the cross-connected charger to be considered "functional," the cross-connected charger must have been surveilled and satisfied the same performance test required for the directly connected charger (i.e., SR 3.8.4.6) within the required Frequency.

In order to limit the time when the DC source is not capable of continuously meeting the single failure criterion, the required charger must be restored to OPERABLE status within 7 days.

The 7 day Completion Time was chosen to allow trouble shooting, location of parts, and repair.

BASES

ACTIONS (continued)

B.1 and B.2

With one battery inoperable, the associated DC system cannot meet its design. It lacks both the surge capacity and the independence from AC power sources which the battery provides if offsite power is lost. Placing the second battery charger in service provides two benefits: 1) restoration of the capacity to supply a sudden DC power demand, and 2) restoration of adequate DC power in the affected train as soon as either AC power distribution system is re-energized following a loss of offsite power. If the cross-connect charger is not placed in service within 2 hours, Condition C would be entered. Additionally for the cross-connected charger to be considered "functional," the cross-connected charger must have been surveilled and satisfied the same performance test required for the directly connected charger (i.e., SR 3.8.4.6) within the required Frequency.

In order to restore the DC source to its design capability, the battery must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is a feature of the original Palisades licensing basis and reflects the availability to provide two trains of DC power from either AC distribution system. Furthermore, it provides a reasonable time to assess plant status as a function of the inoperable DC electrical power source and, if the battery is not restored to OPERABLE status, to prepare to effect an orderly and safe plant shutdown.

C.1 and C.2

If the inoperable DC electrical power source cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous current required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The specified voltage is the nominal rating of the battery. Surveillance voltage measurements may be adjusted for cable losses and for installed plant instrumentation to ensure that battery terminal voltage requirements are satisfied. At that terminal voltage, the battery has sufficient charge to provide the analyzed capacity for either accident loading or station blackout loading. The 7 day Frequency is consistent with manufacturer and IEEE-450 (Ref. 4) recommendations.

SR 3.8.4.2

Visual inspection to detect corrosion of the battery terminals and connectors, or measurement of the resistance of each inter-cell and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The specified limits of ≤ 50 μohm for inter-cell connections and terminal connections, and ≤ 360 μohms for inter-tier and inter-rack connections are in accordance with the manufacturers recommendations. The 50 μohm value is based on the minimum battery design voltage. Battery sizing calculations show the first minute load on the ED-02 battery as the load that determines battery size, hence, battery voltage will be at its lowest value while the battery supplies this current. Calculations also show that at a minimum temperature and end of life (80% battery performance), battery voltage during this first minute load will be about 1.815 V per cell, assuming nominal connection resistance. But if all the connections were at the ceiling value of 50 μohms, the battery manufacturer indicates that the additional voltage drop would result in a battery voltage of about 1.79 V per cell, which is still above the minimum design voltage (Ref. 5).

The 360 μohm value is based on 120% of the nominal cumulative resistance of the components which make up the connections: resistance of the connecting cable, and for each end of the cable, the battery post to cable lug connection, the cable lug itself, and the lug to cable connection.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.2 (continued)

The resistance values determined during initial battery installation are recorded with the battery replacement specifications, FES 95-206-ED-01 and FES 95-206-ED-02.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 12 month Frequency for this SR is consistent with IEEE-450 (Ref. 4), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4.

The specified limits for connection resistance are discussed in the Bases for SR 3.8.4.2.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.4 and SR 3.8.4.5 (continued)

The Surveillance Frequencies of 12 months is consistent with IEEE-450 (Ref. 4), which recommends cell to cell and terminal connection resistance measurement on a yearly basis.

SR 3.8.4.6

This SR requires that each required battery charger be capable of supplying 180 amps at 125 V for ≥ 8 hours. These requirements are based on the design capacity of the chargers. The chargers are rated at 200 amps; the specified 180 amps provides margin between the charger rating and the test requirement.

The specified Frequency requires each required battery charger to be tested each 18 months. The Surveillance Frequency is acceptable, given the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in FSAR Chapter 8 (Ref. 2).

The Surveillance Frequency of 18 months is consistent with the recommendations of RG 1.32 (Ref. 6) and RG 1.129 (Ref. 7), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed 18 months.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.7 (continued)

The reason for the restriction that the plant be outside of MODES 1, 2, 3, and 4 is that performing the Surveillance requires disconnecting the battery from the DC distribution buses and connecting it to a test load resistor bank. This action makes the battery inoperable and completely unavailable for use.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the "as found" condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The modified performance discharge test is a simulated duty cycle that envelopes the Service Test Profile, is approved by the battery manufacturer, and is consistant with IEEE Standards. Since the ampere-hours removed by the initial loads represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capaci[illegible] Initial conditions for the modified performance discharge test shou[illegible] be identical to those specified for a service test.

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

The acceptance criteria for this Surveillance are consistent with the recommendations of IEEE-450 (Ref. 4) and IEEE-485 (Ref. 3). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 4), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 4).

The reason for the restriction that the plant be outside of MODES 1, 2, 3, and 4 is that performing the Surveillance requires disconnecting the battery from the DC distribution buses and connecting it to a test load resistor bank. This action makes the battery inoperable and completely unavailable for use.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17
2. FSAR, Chapter 8
3. IEEE-485-1983, June 1983
4. IEEE-450-1995
5. Letter; Graham Walker, C&D Charter Power Systems, Inc to John Slinkard, Consumers Power Company, 12 July 1996
6. Regulatory Guide 1.32, February 1977
7. Regulatory Guide 1.129, December 1974

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown."

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO requires those, and only those, DC power sources which supply the DC distribution subsystems required by LCO 3.8.10, to be OPERABLE. Each DC source consists of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling. The directly connected battery charger associated with the Diesel Generator (DG) required by LCO 3.8.2, "AC Sources - Shutdown" shall be OPERABLE. This ensures the availability of sufficient DC power sources to maintain the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling).

Two DC sources are required to ensure availability of an uninterruptable power source for the instrumentation and controls for the Reactor Protective System (RPS). If the RPS is not required, then the LCO may continue to be met with one DC source provided:

a. The OPERABLE DC source is associated with the DG required by LCO 3.8.2, "AC Sources - Shutdown", and

b. The cross connected battery charger (powered from the DG required by LCO 3.8.2, "AC Sources - Shutdown") to the alternate bus is OPERABLE and in service.

When an OPERABLE DC electrical power source requires its battery to be OPERABLE, it shall be connected to the associated DC bus. In order for the battery to remain OPERABLE for any extended period of time, at least one charger must be in service. Without a charger in service, the DC loads would reduce the battery charge to the point where the battery would become inoperable. Disconnecting a charger, however, does not, in itself, make a battery inoperable.

BASES

LCO (continued)	The required directly connected charger, ED-15 or ED-16, must be OPERABLE, but need not actually be in service because battery charging current is not needed immediately after an accident.

APPLICABILITY

The DC power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

d. Monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the DC electrical power sources required by this LCO become unavailable during movement of irradiated fuel assemblies.

The DC source requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.4, "DC Sources - Operating."

ACTIONS

A.1

Since the required DC source is only required to support features required by other LCOs, the option to declare those required features with no DC power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, and A.2.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of a required DC source.

Required Actions A.2.1 through A.2.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC sources (and to continue this action until restoration is accomplished) in order to provide the necessary DC power to the plant safety systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required DC power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient control and Preferred AC power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires the SRs from LCO 3.8.4 that are necessary for ensuring the OPERABILITY of the AC sources in MODES 5 and 6.

The SRs from LCO 3.8.4 which are required are those which can be performed without affecting the OPERABILITY or reliability of the required DC source. With only one battery available, loading tests cannot be performed since their performance would render that battery inoperable during the test. This is the case for SRs 3.8.4.7 and 3.8.4.8.

REFERENCES

None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND	This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable for MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating"; during MODES 5 and 6, in the Bases for LCO 3.8.2, "AC Sources - Shutdown." Battery cell parameters satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery cell limits are conservatively established, allowing continued DC electrical system function even when Category A and B limits are not met. The requirement to maintain the average temperature of representative cells above 70°F assures that the battery temperature is within the design band. Battery capacity is a function of battery temperature.
APPLICABILITY	The battery cell parameters are required solely for the support of the associated DC power sources. Therefore, they are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussions in the Bases for LCO 3.8.4 and LCO 3.8.5, "DC Sources - Shutdown."

BASES

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries not within Category A or B limits but within the Category C limits, the battery is not fully charged but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be declared to be inoperable and continued operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells.

Verification that all cells meet the Category C limits (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery will still be capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements may be required to be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A and B limits.

Battery cell parameters must be restored to Category A and B limits within 31 days.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.8.6 A.2 must be initially performed within 24 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 7 days" interval may utilize the 25% SR 3.0.2 extension.

BASES

ACTIONS (continued)

B.1

With the temperature of representative cells below the design temperature, or with one or more battery cells with parameters outside the Category C limits, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable.

Additionally, if battery cells cannot be restored to meeting Category A or B limits within 31 days, a serious difficulty with the battery is indicated and the battery must be declared to be inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 1), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

This Surveillance verification that the average temperature of representative cells is ≥ 70°F is consistent with a recommendation of IEEE-450 (Ref. 1), which states that the temperature of electrolytes in representative cells should be determined on a quarterly basis. The monthly frequency specified is a feature of the initial Palisades license, and is the same as those other pilot cell tests specified in SR 3.8.6.1.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 3.8.6.3

The quarterly inspection of specific gravity and voltage is consistent with the recommendations of IEEE-450 (Ref. 1).

BASES

SURVEILLANCE REQUIREMENTS (continued)

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. Each category is discussed below.

Category A defines the fully charged parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and specific gravity approximate the state of charge of the entire battery.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category A and B limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 1), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A and B limit specified for float voltage is $\geq$ 2.13 V per cell. This value is based on a recommendation of IEEE-450, which states that prolonged operation of cells < 2.13 V can reduce their life expectancy.

The Category A limit specified for specific gravity for each pilot cell is $\geq$ 1.205. This value is six points (0.006) below the average baseline specific gravity for fully charged cells when the battery was installed and is characteristic of a charged cell with adequate capacity. The Category B limit specified for specific gravity for each connected cell is $\geq$ 1.200. Category B also requires that the average of all cells be $\geq$ 1.205 (0.006 below the baseline average of all cells). This allows some cells to be slightly lower than the nominal requirement as long as others are sufficiently higher so as to maintain the average above the nominal full charged value. According to IEEE-450, specific gravity readings are based on a temperature of 77°F (25°C).

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

Category C defines the limit for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limit, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limit specified for electrolyte level (above the top of the plates and not overflowing) ensures that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is based on IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C specific gravity limit that each connected cell must be no less than 0.020 below the average of all connected cells and that average be ≥ 1.195 is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity) (Ref. 2). This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnote (a) allows for the normally observed level increase which occurs during sustained battery charging. Footnotes (b) and (c) to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity readings. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 1).

BASES

REFERENCES

1. IEEE-450-1995
2. C & D Standby Battery Installation and Operation Instructions

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND	The inverters (ED-06, ED-07, ED-08, and ED-09) are the normal source of power for the Preferred AC buses. The function of the inverter is to provide continuous AC electrical power to the Preferred AC buses, even in the event of an interruption to the normal AC power distribution system. A Preferred AC bus can be powered from the AC power distribution system via the Bypass Regulator if its associated inverter is out of service. An interlock prevents supplying more than one Preferred AC bus from the bypass regulator at any time. The station battery provides an uninterruptable power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Features (ESF).
APPLICABLE SAFETY ANALYSES	A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1, "AC Sources - Operating." Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).
LCO	The inverters ensure the availability of Preferred AC power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Maintaining the inverters OPERABLE ensures that the redundancy incorporated into the RPS and ESF instrumentation and controls is maintained. The four inverters ensure an uninterruptable supply of AC electrical power to the Preferred AC buses even if the 2400 V safety related buses are de-energized. An inverter is considered inoperable if it is not powering the associated Preferred AC bus, or if its output voltage or frequency is not within tolerances.

BASES

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that redundant sources of Preferred AC power for instrumentation and control are available to support engineered safeguards equipment in the event of an accident or transient and for power operation, plant heatups and cooldowns, and shutdown operation.

Inverter requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With an inverter inoperable, its associated Preferred AC bus becomes inoperable until it is manually re-energized from the bypass regulator. An inoperable Preferred AC Bus is addressed in LCO 3.8.9.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.7.1 This Surveillance verifies that the inverters are functioning properly and energizing the Preferred AC buses. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESF connected to the Preferred AC buses. The 7 day Frequency takes into account indications available in the control room that alert the operator to inverter malfunctions.
REFERENCES	None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown."

Inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO requires those, and only those, inverters necessary to support the Preferred AC buses required by LCO 3.8.10, "Distribution Systems - Shutdown," to be OPERABLE. As a minimum, both inverters associated with the Diesel Generator (DG) required by LCO 3.8.2, "AC Sources - Shutdown" shall be OPERABLE. This ensures required instrumentation and control functions are maintained by providing an uninterruptable supply of AC electrical power to those Preferred AC buses even if the 2400 V safety related bus is de-energized. If the Reactor Protective System (RPS) is required, then all four inverters shall be OPERABLE.

This ensures the availability of sufficient Preferred AC electrical power to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and loss of shutdown cooling).

An inverter is considered inoperable if it is not powering the associated Preferred AC bus, or if its voltage or frequency is not within tolerances.

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

BASES

APPLICABILITY (continued)

d. Monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the inverters required by this LCO become unavailable during movement of irradiated fuel assemblies.

Inverter requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.7.

ACTIONS

A.1

With a required inverter inoperable, its associated Preferred AC bus becomes inoperable until it is manually re-energized from the bypass regulator. An inoperable Preferred AC Bus is addressed in LCO 3.8.10.

A required inverter would be considered inoperable if it were not available to supply its associated Preferred AC bus. Since the inverter and its associated Preferred AC Bus is only required to support features required by other LCOs, the option to declare those required features without inverter supplied Preferred AC power available to be inoperable, assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, and A.2.4

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.4 provide alternate, but sufficiently conservative, actions for unplanned losses of required inverters.

Required Actions A.2.1 through A.2.4 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters (and to continue this action until restoration is accomplished) in order to provide the required inverter supplied Preferred AC power to the plant instrument and control systems.

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without inverter supplied Preferred AC power.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

A description of the basis for this SR is provided in the Bases for SR 3.8.7.1.

REFERENCES

None

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E AC, DC, and Preferred AC bus electrical power distribution systems are divided into two redundant and independent electrical power distribution trains. Each electrical power distribution train is made up of several subsystems which include the safety related buses, load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.

The Class 1E 2400 V safety related buses, Bus 1C and Bus 1D, are normally powered from offsite, but can be powered from the DGs, as explained in the Background section of the Bases for LCO 3.8.1, "AC Sources - Operating." Each 2400 V safety related bus supplies one train of the Class 1E 480 V distribution system.

The 120 V Preferred AC buses are normally powered from the inverters. The alternate power supply for the buses is a constant voltage transformer, called the Bypass Regulator. Use of the Bypass regulator is governed by LCO 3.8.7, "Inverters - Operating." The bypass regulator is powered from the non-Class 1E instrument AC bus, Y-01. The Instrument AC bus is normally powered through an automatic bus transfer switch, an instrument AC transformer, and isolation fuses. Its normal power source is MCC-1. Loss of power to MCC-1 will cause automatic transfer of the Instrument AC bus to MCC-2.

There are two independent 125 V DC electrical power distribution subsystems.

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable in MODES 1, 2, 3, and 4 is provided in the Bases for LCO 3.8.1.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The AC, DC, and Preferred AC bus electrical power distribution subsystems are required to be OPERABLE. The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and Preferred AC bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA.

BASES

LCO (continued)

Maintaining both trains of AC, DC, and Preferred AC bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the plant design is not defeated. Therefore, a single failure within any electrical power distribution subsystem will not prevent safe shutdown of the reactor.

OPERABLE electrical power distribution subsystems require the buses, load centers, motor control centers, and distribution panels listed in Table B 3.8.9-1 to be energized to their proper voltages. In addition, tie breakers between redundant safety related AC power distribution subsystems must be open when a 2400 V source is OPERABLE for each train. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem. If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 2400 V buses from being powered from the same offsite circuit or preclude cross connecting Class 1E 480 V subsystems when 2400 V power is available for only one train.

This LCO does not address the power source for the Preferred AC buses. The Preferred AC buses are normally powered from the associated inverter. An alternate source, the Bypass Regulator, is available to supply one Preferred bus at a time, to allow maintenance on an inverter. The proper alignment of the inverter output breakers is addressed under the inverter LCOs. Therefore a Preferred AC Bus may be considered OPERABLE when powered from either the associated inverter or the Bypass Regulator as long as the voltage and frequency of the supply is correct.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that AC, DC, and Preferred AC power is available to the redundant trains and channels of safeguards equipment, instrumentation and controls required to support engineered safeguards equipment in the event of an accident or transient.

Electrical power distribution subsystem requirements for MODES 5 and 6, and during movement of irradiated fuel assemblies are addressed in LCO 3.8.10, "Distribution Systems - Shutdown."

BASES

ACTIONS

A.1

With one or more required AC buses, load centers, motor control centers, or distribution panels, except Preferred AC buses, in one train inoperable, the redundant AC electrical power distribution subsystem in the other train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because an additional failure in the power distribution systems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combinations of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one Preferred AC bus inoperable, the remaining OPERABLE Preferred AC buses are capable of supporting the minimum safety functions necessary to shut down the plant and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the Preferred AC bus must be restored to OPERABLE status within 8 hours by powering it from the associated inverter or from the Bypass Regulator.

BASES

ACTIONS

B.1 (continued)

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate Preferred AC power and is a feature of the original Palisades licensing basis.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the Preferred AC distribution system. At this time, a DC bus could again become inoperable, and Preferred AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one or more DC bus in one train inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the required DC buses must be restored to OPERABLE status within 8 hours by powering the bus from the associated battery or charger.

This 8 hour limit is more conservative than Completion Times allowed for the vast majority of components which would be without power and is a feature of the original Palisades licensing basis.

BASES

ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single continuous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 16 hours, since initial failure of the LCO, to restore the Preferred DC distribution system. At this time, a AC bus could again become inoperable, and Preferred AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E corresponds to a degradation in the electrical distribution system that causes a required safety function to be lost. When more than one Condition is entered, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This surveillance verifies that the required AC, DC, and Preferred AC bus electrical power distribution subsystems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained.

For those buses which have undervoltage alarms in the control room, correct voltage may be verified by the absence of an undervoltage alarm.

For those buses which have only one possible power source and have undervoltage alarms in the control room, correct breaker alignment may be verified by the absence of an undervoltage alarm.

A Preferred AC Bus may be considered correctly aligned when powered from either the associated inverter or from the bypass regulator. A mechanical interlock prevents connecting two or more Preferred AC Buses to the Bypass Regulator. LCO 3.8.7 and LCO 3.8.8 address the condition of supplying a Preferred AC Bus from the bypass regulator.

The 7 day Frequency takes into account the redundant capability of the AC, DC, and Preferred AC bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

None

TABLE B 3.8.9-1 (page 1 of 1)
Required Electrical Distribution Trains

TYPE	VOLTAGE	LEFT TRAIN	RIGHT TRAIN
AC Power Distribution Subsystems	2400	Bus 1C	Bus 1D
	480	Bus 11	Bus 12
	480	Bus 19	Bus 20
	480	MCC 1	MCC 2
	480	MCC 7	MCC 8
	480	MCC 21	MCC 22
	480	MCC 23	MCC 24
	480	MCC 25	MCC 26
DC Power Distribution Subsystems	125	Bus D10-L	Bus D20-L
	125	Bus D10-R	Bus D20-R
	125	Pnl D11A	Pnl D21A
	125	Pnl D11-1	Pnl D21-1
	125	Pnl D11-2	Pnl D21-2
Preferred AC Subsystems	120	Bus Y-10	Bus Y-20
	120	Bus Y-30	Bus Y-40

3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and Preferred AC bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

A description of the Safety Analyses applicable during MODES 5 and 6 is provided in the Bases for LCO 3.8.2, "AC Sources - Shutdown."

The distribution system satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO requires those, and only those, AC, DC, and Preferred AC distribution subsystems to be OPERABLE which are necessary to support equipment required by other LCOs.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

This LCO does not address the power source for the Preferred AC buses. The Preferred AC buses are normally powered from the associated inverter. An alternate source, the Bypass Regulator, is available to supply one Preferred bus at a time, to allow maintenance on an inverter. The proper alignment of the inverter output breakers is addressed under LCO 3.8.8, "Inverters - Shutdown." Therefore a Preferred AC Bus may be considered OPERABLE when powered from either the associated inverter or the Bypass Regulator as long as the voltage and frequency of the supply is correct.

APPLICABILITY

The electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that equipment and instrumentation is available to:

a. Provide coolant inventory makeup,

b. Mitigate a fuel handling accident,

c. Mitigate shutdown events that can lead to core damage, and

BASES

APPLICABILITY (continued)

d. Monitoring and maintaining the plant in a cold shutdown condition and refueling condition.

This LCO is applicable during movement of irradiated fuel assemblies even if the plant is in a condition other than MODE 5 or 6. This LCO provides the necessary ACTIONS if the electrical power distribution subsystems required by this LCO become unavailable during movement of irradiated fuel assemblies.

The electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are addressed in LCO 3.8.9, "Distribution Systems - Operating."

ACTIONS

A.1

Since the distribution systems are only required to support features required by other LCOs, the option to declare those affected required features to be inoperable assures that appropriate ACTIONS will be implemented in accordance with the affected LCOs.

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Required Action A.1 may involve undesired and unnecessary administrative efforts, therefore, Required Actions A.2.1 through A.2.5 provide alternate, but sufficiently conservative, actions for unplanned losses of power to distribution systems.

Required Actions A.2.1, A.2.2, A.2.3, and A.2.5 require suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions, and declaration that affected shutdown cooling trains are inoperable. The suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies does not preclude actions to place a fuel assembly in a safe location; the suspension of positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SHUTDOWN MARGIN is maintained.

These ACTIONS minimize the probability or the occurrence of postulated events. It is further required (Required Action A.2.4) to immediately initiate action to restore the required distribution subsystems (and to continue this action until restoration is accomplished) in order to provide the necessary electrical power to the plant safety systems.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

The Completion Time of "immediately" is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

A description of the basis for this SR is provided in the Bases for SR 3.8.9.1.

REFERENCES

None

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Primary Coolant System (PCS), and refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. The refueling operations boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The REFUELING BORON CONCENTRATION limit is defined in Section 1.1, "Definitions." Plant procedures ensure the specified boron concentration in order to maintain the reactor core subcritical by at least 5% $\Delta\rho$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures. During evolutions where plant procedures allow manipulation of control rods or where conditions could result in inadvertent control rod withdrawal, such as reactor vessel head removal, the boron concentration must be sufficient to assure that the reactor core will remain subcritical by at least 5% $\Delta\rho$ without taking credit for the negative reactivity provided by the control rods (i.e., assuming all rods fully withdrawn). During evolutions where the control rods are inserted, plant procedures do not allow manipulation of control rods, and conditions do not exist that could result in inadvertent rod withdrawal, such as MODE 6 operations with the Upper Guide Structure in place (other than during head removal). Therefore, credit may be taken for the negative reactivity provided by the control rods when determining the boron concentration necessary to assure that the reactor core will remain subcritical by at least 5% $\Delta\rho$.

The Palisades Nuclear Plant design criteria requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) System is capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

BASES

BACKGROUND (continued)

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the PCS is cooled and depressurized the vessel head is unbolted and the head is removed. The refueling cavity is then flooded with borated water from the safety injection refueling water tank into the open reactor vessel by gravity feeding or by the use of the spent fuel cooling, safety injection pumps, or charging pumps.

The pumping action of the SDC System in the PCS and the natural circulation due to thermal driving head in the reactor vessel mix the added concentrated boric acid with the water in the refueling cavity. The SDC System is in operation during refueling (see LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level") to provide forced circulation in the PCS and to assist in maintaining the REFUELING BORON CONCENTRATION in the PCS, and the refueling cavity.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident analysis and is conservative for MODE 6. The REFUELING BORON CONCENTRATION limit is based on the core reactivity and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that demonstrate the correct fuel loading plan (including full core mapping) ensure the reactor core will remain subcritical by at least 5% $\Delta\rho$ during the refueling operation.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling cavity, and the reactor vessel form a single connected water mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in B 3.1.1, "SHUTDOWN MARGIN."

Boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2).

BASES

LCO

The LCO requires that a minimum boron concentration be maintained in the PCS, and refueling cavity while in MODE 6. The boron concentration limit specified ensures the reactor core will remain subcritical by at least 5% $\Delta\rho$ during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures the reactor core will remain subcritical by at least 5% $\Delta\rho$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensures that an adequate amount of negative reactivity is available to shut down the reactor and to maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the plant in compliance with the LCO. If the boron concentration of any coolant volume in the PCS or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position, or normal cooldown of the coolant volume for the purpose of system temperature control.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for plant conditions.

BASES

ACTIONS	A.3 (continued) Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.
SURVEILLANCE REQUIREMENTS	SR 3.9.1.1 This SR ensures the coolant boron concentration in the PCS and the refueling cavity is within the limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis. A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.
REFERENCES	1. FSAR, Section 5.1 2. FSAR, Section 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range channels (NI-1/3 and NI-2/4) are used during refueling operations to monitor the core reactivity condition. The installed source range channels are part of the Nuclear Instrumentation System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core. The use of portable detectors in place of installed detectors is permitted, provided the LCO requirements are met.

The installed source range channels utilize fission detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers five decades of neutron flux (1E+5 cps). The detectors provide continuous visual and audible indication in the control room to alert operators to a possible dilution accident. The Nuclear Instrumentation System is designed in accordance with the criteria presented in Reference 1.

If used, portable detectors should be functionally equivalent to the installed source range channels.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range channels are required to provide a signal to alert the operator to unexpected changes in core reactivity such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. The analysis of the uncontrolled boron dilution accident shows that normally available SHUTDOWN MARGIN would be reduced, but there is sufficient time for the operator to take corrective actions.

Nuclear Instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

This LCO requires two source range channels OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each channel must provide visual indication and at least one of the two channels must provide an audible count rate function in the control room. Therefore, with no audible count rate function from at least one channel, both source range channels would be inoperable.

BASES

APPLICABILITY

In MODE 6, the source range channels must be OPERABLE to detect changes in core reactivity. There is no other direct means available to check core reactivity levels.

In MODES 3, 4, and 5, the installed source range channels are required to be OPERABLE by LCO 3.3.9, "Neutron Flux Monitoring Channels." In MODES 1, 2, and 3, one source range channel is required by LCO 3.3.8, "Alternate Shutdown System."

ACTIONS

A.1 and A.2

With only one source range channel OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

B.1

With no source range channel OPERABLE, action to restore a channel to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until one source range channel is restored to OPERABLE status.

B.2

With no source range channel OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range channel are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to verify that the required boron concentration exists. The Completion Time of once per 12 hours is sufficient to obtain and analyze a primary coolant and refueling cavity sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this period.

BASES

ACTIONS

B.2 (continued)

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." . . .however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. . . .Therefore, while Required Action 3.9.2 B.2 must be initially performed within 12 hours without any SR 3.0.2 extension, subsequent performances may utilize the 25% SR 3.0.2 extension.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions, but does not require the two source range channels to have the same reading. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.9.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. FSAR, Section 7.6
2. FSAR, Section 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, a release of fission product radioactivity within the containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are filtered, closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50, Appendix J leakage criteria and tests are not required. In MODE 5, no accidents are assumed which will result in a release of radioactive material to the containment atmosphere. Therefore, no requirements are stipulated for containment penetrations in MODE 5.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment structure provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with the equipment hatch closed, the hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with the equipment hatch removed, the OPERABILITY requirements of the Fuel Handling Area Ventilation System must be met. These OPERABILITY requirements are provided in LCO 3.7.12, "Fuel Handling Area Ventilation System."

BASES

BACKGROUND (continued)

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed. An exception, however, is provided for the personnel air lock. It is acceptable to have both doors of the personnel air lock open simultaneously provided the equipment hatch is open.

The requirements on containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Vent System includes a 12 inch purge penetration and two 8 inch exhaust penetrations. During MODES 1, 2, 3, and 4, the valves in the purge and vent penetrations are secured in the closed position and venting the containment is accomplished using the Clean Waste Receiving Tank (CWRT) vent line. The two valves in the CWRT vent line penetration are closed automatically by a Containment High Radiation signal. Neither the Containment Purge and Vent System, nor the CWRT vent line is subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The Purge and Vent System is used for this purpose. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment with either the Containment Purge and Vent System in operation, or the CWRT aligned for containment venting, the associated isolation valves must be capable of being closed by an OPERABLE channel of radiation instrumentation required by LCO 3.3.6, "Refueling Containment High Radiation Instrumentation."

BASES

BACKGROUND (continued)	Other containment penetrations that provide direct access from containment atmosphere to outside atmosphere that are not capable of being closed by an OPERABLE Refueling Containment High Radiation signal must be isolated on at least one side. Containment penetrations "that provide direct access from containment atmosphere to outside atmosphere" are those which would allow passage of air containing radioactive particulates to migrate from inside the containment to the atmosphere outside the containment even though no measurable differential pressure existed. Specifically, they do not include penetrations which are filtered, or penetrations whose piping is filled with liquid. Isolation may be achieved by a manual or automatic isolation valve, blind flange, or equivalent. Equivalent isolation methods, authorized under the provisions of 10 CFR 50.59, may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements.
APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). The requirements of LCO 3.9.6, "Refueling Cavity Water Level," (and the minimum decay time of 48 hours required by the Operating Requirements Manual) prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are less than the guideline values specified in 10 CFR 100. Containment penetration isolation is not required by the fuel handling accident to maintain offsite doses within the guidelines of 10 CFR 100, but operating experience indicates that containment isolation provides significant reduction of the resulting offsite doses. Therefore, the Containment Penetrations satisfy the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
LCO	This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment hatch, air locks and any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment penetrations.

BASES

LCO (continued)	For the OPERABLE containment penetrations, this LCO ensures that these penetrations are isolable by the Refueling Containment High Radiation instrumentation. The OPERABILITY requirements for this LCO do not assume a specific closure time for the valves in these penetrations since the accident analysis makes no specific assumptions about containment closure time after a fuel handling accident. LCO 3.9.3.a is modified by a Note which allows the equipment hatch to be opened if the Fuel Handling Area Ventilation System is in compliance with LCO 3.7.12. LCO 3.9.3.b is modified by a Note which allows both doors of the personnel air lock to be simultaneously opened provided the equipment hatch is opened. In the event of a fuel handling accident inside containment with both doors in the personnel air lock open and the equipment hatch open, the Fuel Handling Area Ventilation System would be available to filter the fission products in the containment atmosphere prior to their being released to the environment and thereby significantly reducing the offsite dose.
APPLICABILITY	The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.
ACTIONS	<u>A.1 and A.2</u> With the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including the Refueling Containment High Radiation instrumentation not being capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition in which containment closure is not needed.

BASES

ACTIONS

A.1 and A.2 (continued)

This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the valves in unisolated penetrations which provide a direct path from the containment atmosphere to the outside atmosphere will demonstrate that the valves are not blocked from closing. Also, the Surveillance will demonstrate that each valve operator has motive power, which will ensure each valve is capable of being closed by an OPERABLE Refueling Containment High Radiation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance provides assurance that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in an excessive release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each automatic isolation valve providing direct access from the containment atmosphere to the outside atmosphere valve actuates to its isolation position on an actual or simulated high radiation signal.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.3.2 (continued) The SR is modified by a Note which requires only the valves in unisolated penetrations to be tested. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. LCO 3.3.6, "Refueling Containment High Radiation Instrumentation," requires a CHANNEL CHECK every 7 days, a CHANNEL FUNCTIONAL TEST every 31 days and a CHANNEL CALIBRATION every 18 months to ensure the channel OPERABILITY during refueling operations. These surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.
REFERENCES	1. FSAR, Section 14.19

B 3.9 REFUELING OPERATIONS

B 3.9.4 Shutdown Cooling (SDC) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Primary Coolant System (PCS) as required by the Palisade Nuclear Plant design, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the PCS by circulating primary coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the PCS via the PCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of primary coolant through the SDC heat exchanger(s). Mixing of the primary coolant is maintained by this continuous circulation of primary coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the primary coolant temperature is not maintained below 200°F, boiling of the primary coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the primary coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical.

The loss of primary coolant and the reduction of boron concentration in the primary coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the SDC System is required to be in operation in MODE 6, with the refueling cavity water level greater than or equal to the 647 ft elevation, to prevent this challenge. The LCO allows the removal of an SDC train from operation for short durations under the condition that the boron concentration of the primary coolant is not reduced.

This conditional allowance does not result in a challenge to the fission product barrier.

SDC and Coolant Circulation - High Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

Only one SDC train is required for decay heat removal in MODE 6, with the refueling cavity water level greater than or equal to the 647 ft elevation. Only one SDC train is required because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one SDC train must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE SDC train consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the PCS temperature. The flow path starts in one of the PCS hot legs and is returned to at least one PCS cold leg.

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

The LCO is modified by two Notes. Note 1 allows the required operating SDC train to not be in operation for up to 1 hour in each 8 hour period, provided no operations are permitted that would cause a reduction of the PCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles, and PCS to SDC isolation valve testing.

During this 1 hour period, decay heat is removed by natural circulation to the large mass of water in the refueling cavity. Note 2 allows the required SDC train to be made inoperable for ≤ 2 hours per 8 hour period for testing and maintenance provided one SDC train in operation providing flow through the reactor core, and the core outlet temperature is ≤ 200°F. The purpose of this Note is to allow the heat flow path from the SDC heat exchanger to be temporarily interrupted for maintenance or testing on the Component Cooling Water or Service Water Systems.

BASES

LCO (continued)

During this 2 hour period, the core outlet temperature must be maintained ≤ 200°F. Requiring one SDC train to be in operation ensures adequate mixing of the borated coolant.

APPLICABILITY

One SDC train must be OPERABLE and in operation in MODE 6, with the refueling cavity water level greater than or equal to 647 ft elevation, to provide decay heat removal. The 647 ft elevation was selected because it corresponds to the elevation requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, "Primary Coolant System (PCS)." SDC train requirements in MODE 6, with the refueling cavity water level less than the 647 ft elevation are located in LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level."

ACTIONS

SDC train requirements are met by having one SDC train OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If one required SDC train is inoperable or not in operation, actions shall be immediately initiated and continued until the SDC train is restored to OPERABLE status and to operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

A.2

If SDC train requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur through the addition of water with a lower boron concentration than that contained in the PCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

BASES

ACTIONS (continued)

A.3

If SDC train requirements are not met, actions shall be taken immediately to suspend loading irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural circulation to the heat sink provided by the water above the core. A minimum refueling cavity water level equivalent to the 647 ft elevation provides an adequate available heat sink. Suspending any operation that would increase the decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.4

If SDC train requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed to prevent fission products, if released by a loss of decay heat removal event, from escaping to the environment. The 4 hour Completion Time is based on the low probability of the coolant boiling in that time and allows time for fixing most SDC problems.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the SDC train is in operation and circulating primary coolant. The flow rate is sufficient to provide decay heat removal capability and to prevent thermal and boron stratification in the core. The 1000 gpm flow rate has been determined by operating experience rather than analysis. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the SDC System.

REFERENCES

1. FSAR, Sections 6.1 and 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.5 Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purposes of the SDC System in MODE 6 are to remove decay heat and sensible heat from the Primary Coolant System (PCS), as required by the Palisades Nuclear Plant design, to provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification (Ref. 1). Heat is removed from the PCS by circulating primary coolant through the SDC heat exchanger(s), where the heat is transferred to the Component Cooling Water System via the SDC heat exchanger(s). The coolant is then returned to the PCS via the PCS cold leg(s). Operation of the SDC System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of primary coolant through the SDC heat exchanger(s) and bypassing the heat exchanger(s). Mixing of the primary coolant is maintained by this continuous circulation of primary coolant through the SDC System.

APPLICABLE SAFETY ANALYSES

If the primary coolant temperature is not maintained below 200°F, boiling of the primary coolant could result. This could lead to inadequate cooling of the reactor fuel due to the resulting loss of coolant in the reactor vessel. Additionally, boiling of the primary coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity, and because of the possible addition of water to the reactor vessel with a lower boron concentration than is required to keep the reactor subcritical.

The loss of primary coolant and the reduction of boron concentration in the primary coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the SDC System are required to be OPERABLE, and one train is required to be in operation in MODE 6, with the refueling cavity water level less than the 647 ft elevation to prevent this challenge.

SDC and Coolant Circulation - Low Water Level satisfies Criterion 4 of 10 CFR 50.36(c)(2).

BASES

LCO

In MODE 6, with the refueling cavity water level less than the 647 ft elevation, both SDC trains must be OPERABLE. Additionally, one train of the SDC System must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of a criticality; and

c. Indication of primary coolant temperature.

An OPERABLE SDC train consists of an SDC pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the PCS temperature. The flow path starts in one of the PCS hot legs and is returned to the PCS cold legs.

A SDC train may be considered OPERABLE (but not necessarily in operation) during re-alignment to, and when it is re-aligned for, LPSI service or for testing, if it is capable of being (locally or remotely) realigned to the SDC mode of operation and is not otherwise inoperable. Since SDC is a manually initiated system, it need not be considered inoperable solely because some additional manual valve realignments must be made in addition to the normal initiation actions. Because of the dual functions of the components that comprise the LPSI and shutdown cooling systems, the LPSI alignment may be preferred.

Both SDC pumps may be aligned to the safety injection refueling water tank to support filling the refueling cavity or for performance of required testing.

APPLICABILITY

Two SDC trains are required to be OPERABLE, and one SDC train must be in operation in MODE 6, with the refueling cavity water level less than the 647 ft elevation to provide decay heat removal. Requirements for the SDC System in other MODES are covered by LCOs in Section 3.4, "Primary Coolant System." MODE 6 requirements, with the refueling cavity water level greater than or equal to the 647 ft elevation are covered in LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level."

BASES

ACTIONS

A.1 and A.2

If one SDC train is inoperable, action shall be immediately initiated and continued until the SDC train is restored to OPERABLE status, or until a water level of greater than or equal to the 647 ft elevation is established. When the water level is established at the 647 ft elevation or greater, the plant conditions will change so that LCO 3.9.4, "Shutdown Cooling and Coolant Circulation - High Water Level," is applicable, and only one SDC train is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no SDC train is in operation or no SDC trains are OPERABLE, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations can occur by the addition of water with lower boron concentration than that contained in the PCS. Therefore, actions that reduce boron concentration shall be suspended immediately.

B.2

If no SDC train is in operation or no SDC trains are OPERABLE, action shall be initiated immediately and continued without interruption to restore one SDC train to OPERABLE status and operation. Since the plant is in Conditions A and B concurrently, the restoration of two OPERABLE SDC trains and one operating SDC train should be accomplished expeditiously.

B.3

If no SDC train is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed Immediately. With the SDC train requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one SDC train is operating and circulating primary coolant. The flow rate is sufficient to provide decay heat removal capability and to prevent thermal and boron stratification in the core.

In addition, during operation of the SDC train with the water level in the vicinity of the reactor vessel nozzles, the SDC train flow rate determination must also consider the SDC pump suction requirements. The 1000 gpm flow rate has been determined by operating experience rather than analysis. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator to monitor the SDC System in the control room.

SR 3.9.5.2

Verification that the required pump is OPERABLE ensures that an additional SDC pump can be placed in operation, if needed, to maintain decay heat removal and primary coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Sections 6.1 and 14.3

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND

The performance of CORE ALTERATION or the movement of irradiated fuel assemblies within containment requires a minimum water level greater than or equal to the 647 ft elevation. During refueling this maintains sufficient water level in the refueling cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than the guidelines of 10 CFR 100.

APPLICABLE SAFETY ANALYSES

During core alterations and during movement of irradiated fuel assemblies, the water level in the refueling cavity is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Regulatory Guide (RG) 1.25 (Ref. 1). The fuel handling accident analysis inside containment is described in Reference 2.

A minimum water level of 647 feet provides 22.5 feet of water above the refueling cavity floor. RG 1.25 position C.1.g specifies a decontamination factor of 100 for a water depth of 23 feet above the failed fuel; footnote d, however, provides the following guideline for situations where water depths are less than 23 feet:

> *". . . for a water depth of less than 23 ft., the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable in conservatism to those of this guide."*

Since RG 1.25 does not give specific guidance reguarding the assumed Iodine removal mechanisms in the water, a linear relationship between the decontamination factor and the water depth was assumed. The resulting decontamination factor for 22.5 feet of water is 97.43. Therefore, assuming the following:

BASES

APPLICABLE SAFETY ANALYSIS (continued)

A decontamination factor of 97.43,

A minimum decay time of 48 hours between power operation and commencement of fuel handling, and 12% of the total fuel rod iodine 131 inventory is contained in the fuel pellet to cladding gap;

the analyses in Reference 2 demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within the guidelines of 10 CFR 100.

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

A minimum refueling cavity water level greater than or equal to the 647 ft elevation is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are less than the guideline of 10 CFR 100.

APPLICABILITY

LCO 3.9.6 is applicable during CORE ALTERATIONS, and when moving fuel assemblies in the presence of irradiated fuel assemblies in containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Spent Fuel Pool Water Level."

ACTIONS

A.1 and A.2

With a water level below the 647 ft elevation, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 Verification of a minimum water level corresponding to the 647 ft elevation ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required elevation limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.25, March 23, 1972 2. FSAR, Section 14.19