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Perry Nuclear Power Plant
Docket No. 50-440
Submittal of Revision 2 to the Technical Specification Bases

Ladies and Gentlemen:

Pursuant to the requirements of Technical Specification Section 5.5.11.d, the Perry Nuclear Power Plant (PNPP) is submitting a copy of Revision 2 of the Technical Specification Bases. This submittal reflects changes made since the submittal of Revision 1 of the Technical Specification Bases (April 4, 1998). The changes are identified by sidebars.

If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-6305.

Very truly yours,

John K. Wood

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

A001

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

To meet General Design Criterion 26, the SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. In addition, the SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

The SLC System consists of a boron solution storage tank, two positive displacement pumps and two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the control room, as directed by the Plant Emergency Instructions, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration of at least 816 ppm of natural boron in the reactor core at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). The concentration versus volume limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

pump suction low level trip in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

Credit is also taken for use of the SLC System in the licensing basis radiological calculations for an accident that causes severe core damage. In such an event, radioactive iodine (and other fission products other than the noble gases) would be released into the drywell and containment atmosphere primarily in an aerosol (particulate) form, such as cesium iodide and other iodide forms. These iodide compounds will settle out, or be sprayed out of the containment atmosphere by the RHR Containment Sprays. Without any pH control of the water in the reactor vessel and suppression pool, large fractions of the dissolved iodides could be converted to elemental iodine and be re-evolved into the containment atmosphere. However, if the pH is controlled and maintained at a value of 7 or greater, very little (less than 1%) of the dissolved iodides will be converted and re-evolved. Therefore, the SLC system, which contains a buffering solution that raises the pH of water, would be injected. This solution would mix in the reactor vessel and suppression pool water, and ensure that long-term vessel and suppression pool pH levels are maintained greater than or equal to 7.

The SLC System satisfies the requirements of the NRC Policy Statement because operating experience and probabilistic risk assessment have generally shown it to be important to public health and safety.

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in the shutdown position and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE during these conditions, when only a single control rod can be withdrawn.

ACTIONS

A.1

With one SLC subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an

(continued)

BASES

ACTIONS

A.1 (continued)

OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or transient occurring concurrent with the failure of the Control Rod Drive System to shut down the reactor.

B.1

With two SLC subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, based on the low probability of a DBA or transient occurring concurrent with the failure of the Control Rod Drive System to shut down the reactor.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances, verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borax-boric acid solution in the storage tank, and temperature of the pump suction piping), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System flow path ensures that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position, provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve controls. This is acceptable since the SLC System is a manually initiated system. This Surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

SR 3.1.7.5

This Surveillance requires an examination of the borax-boric acid solution by using chemical analysis to ensure the proper concentration of boron exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to establish that the boron solution concentration is within the specified limits. This Surveillance must be performed anytime the solution temperature is restored to ≥ 70°F, to ensure no significant

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.5 (continued)

boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating each SLC System pump develops a flow rate ≥ 32.4 gpm at a discharge pressure ≥ 1220 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the borax-boric acid solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months, at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance was performed with the reactor at power. Operating experience

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power-High (continued)

Six channels of Average Power Range Monitor Flow Biased Simulated Thermal Power–High, with three channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives one total drive flow signal representative of total core flow. The recirculation loop drive flow signals are generated by eight flow units. One flow unit from each recirculation loop is provided to each APRM channel. Total drive flow is determined by each APRM by summing up the flow signals provided to the APRM from the two recirculation loops.

The clamped Allowable Value function was not specifically credited in the accident analysis, but it is retained for RPS as required by the NRC approved licensing basis. The THERMAL POWER time constant provided in the CORE OPERATING LIMITS REPORT is representative of the fuel heat transfer dynamics and provides a signal that is proportional to the THERMAL POWER.

The Average Power Range Monitor Flow Biased Simulated Thermal Power–High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux–High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux-High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 2, the Average Power Range Monitor Fixed Neutron Flux–High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.5

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criteria are included in Reference 5.

A Note to the Surveillance states that breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. This is allowed since the arc suppression time is short and does not appreciably change.

EOC-RPT SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 18 months. The frequency is based upon plant operating experience, which shows that random failures of instrumentation components that cause serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.4.1.6

This SR ensures that the RPT breaker arc suppression time is provided to the EOC-RPT SYSTEM RESPONSE TIME test. The 60 month Frequency of the testing is based on the difficulty of performing the test and the reliability of the circuit breakers.

REFERENCES

1. USAR, Section 7.6.1.6.

2. USAR, Section 5.2.2.

3. USAR, Sections 15.1.1, 15.1.2, and 15.1.3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a, 2.e. Reactor Vessel Water Level–Low Low, Level 2 (continued)

since isolation of these valves is not critical to orderly plant shutdown.

This Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.
However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE.

This Function isolates the 1E22-F023 Valve (Function 2.e), and the Group 1, 5, 7, and 8 valves (Function 2.a).

2.b, 2.d, 2.f Drywell Pressure–High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure–High Function associated with isolation of the primary containment is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA. In addition, Functions 2.b and 2.d provide isolation signals to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the drywell suppression function of the drywell.

High drywell pressure signals are initiated from four pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure–High per Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Function 2.f (Division 3) has only one trip system consisting of four channels logically combined in a one-out-of-two twice configuration.

The Allowable Value was selected to be the same as the ECCS Drywell Pressure–High Allowable Value (LCO 3.3.5.1), since

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Reactor Vessel Water Level–Low Low Low, Level 1 (continued)

This Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs. However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE.

This Function isolates the Group 2 isolation valves.

2.g. Containment and Drywell Purge Exhaust–Plenum Radiation–High

High purge exhaust plenum radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from four the primary containment due to a break in the RCPB. When Purge Exhaust-Plenum Radiation–High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products. Additionally, the Purge Exhaust-Plenum Radiation–High is assumed to initiate isolation of the primary containment during a fuel handling accident involving handling of recently irradiated fuel (Ref. 2). In addition, this Function provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the drywell suppression function of the drywell.

The Purge Exhaust-Plenum Radiation–High signals are initiated from four radiation detectors that are located on the purge exhaust plenum ductwork coming from the drywell and containment. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.g. Containment and Drywell Purge Exhaust-Plenum Radiation - High (continued)

Four channels of Containment and Drywell Purge Exhaust-Plenum Radiation-High Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Containment and Drywell Purge System inboard and outboard isolation valves each use a separate two-out-of-two isolation logic.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits.

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs), and movement of recently irradiated fuel assemblies in the primary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure offsite dose limits are not exceeded. However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

These Functions isolate the Group 8 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h Manual Initiation (continued)

Four channels of the Manual Initiation Function are required to be OPERABLE in MODES 1, 2, and 3, and during movement of recently irradiated fuel assemblies in primary containment, or operations with a potential for draining the reactor vessel, since these are the MODES in which the Primary Containment and Drywell Isolation automatic Functions are required to be OPERABLE. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE.

The manual initiation channels for the RCIC System is discussed in Section 3.k below, and for the HPCS System is discussed in the Bases description for ECCS Instrumentation (LCO 3.3.5.1).

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow-High

RCIC Steam Line Flow-High Function is provided to detect a break of the RCIC steam lines and initiates closure of the steam line isolation valves. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and core uncovery can occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for this Function is not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. However, these instruments prevent the RCIC steam line break from becoming bounding.

The RCIC Steam Line Flow-High signals are initiated from two transmitters that are connected to the system steam lines. Two channels of RCIC Steam Line Flow-High Functions are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.a. RCIC Steam Line Flow-High (continued)

The Allowable Value is chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

(continued)

BASES

ACTIONS (continued)

K.1, K.2.1 and K.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be isolated (Required Action K.1). Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable instrumentation. Alternately, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission production release. Actions must continue until OPDRVs are suspended.

L.1

If applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Primary Containment and Drywell Isolation Instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains primary containment isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure–High (continued)

Drywell Pressure–High signals are initiated from four pressure transmitters that sense drywell pressure. Four channels of Drywell Pressure–High Function (two channels per trip system) are required to be OPERABLE to ensure that no single instrument failure can preclude CRER System initiation.

The Drywell Pressure–High Allowable Value was chosen to be the same as the ECCS Drywell Pressure–High Allowable Value (LCO 3.3.5.1).

The Drywell Pressure–High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure–High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure–High setpoint.

3. Control Room Ventilation Radiation Monitor

The Control Room Ventilation Radiation Monitor measures radiation levels downstream of the supply plenum discharge of the control room. A high radiation level may pose a threat to control room personnel; thus, the Control Room Ventilation Radiation Monitor Function will automatically initiate the CRER System.

The Control Room Ventilation Radiation Monitor Function consists of one noble gas monitor. One channel (which provides input to both Trip Systems) of the Control Room Ventilation Radiation Monitor is required to be OPERABLE. Since a LOCA signal will also initiate the CRER System isolating the control room from the environment, and considering the fact that a LOCA signal itself incorporates sufficient redundancy, the airborne radiation monitor signal is considered a diverse signal, and does not require redundancy. The Allowable Value was selected to ensure protection of the control room personnel.

The Control Room Ventilation Radiation Monitor Function is required to be OPERABLE in MODES 1, 2, and 3, and during OPDRVs and movement of recently irradiated fuel in the primary containment or Fuel Handling Building to ensure

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Control Room Ventilation Radiation Monitor (continued)

that control room personnel are protected during a LOCA, fuel handling event, or a vessel draindown event. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or fuel damage is low; thus, the Function is not required.

ACTIONS

A Note has been provided to modify the ACTIONS related to CRER System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CRER System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CRER System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

Because of the diversity of sensors available to provide initiation signals and the redundancy of the CRER System design, an allowable out of service time of 24 hours has been shown to be acceptable (Refs. 4 and 5) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable

(continued)

BASES

LCO (continued)

below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for conditions. The transient evaluations in Reference 3 are based on these setpoints, but also include the additional uncertainties of ± 3% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism. Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test, and in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

The Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings. The safety lift setpoints will still be set within a tolerance of ± 1%, but the setpoints will be tested to within ± 3% to determine acceptance or failure of the as-found valve lift setpoint (Reference 4).

SR 3.4.4.2

The required relief function S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions i.e., solenoids of the automatic relief function operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.4 overlaps this SR to provide complete testing of the safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.4.3

A manual actuation of each required S/RV is performed to verify that the valve is functioning properly and that no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, by the S/RV discharge pipe pressure switch, or any other method suitable to verify steam flow (e.g., tailpipe temperature.)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is the pressure recommended by the valve manufacturer. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. SR 3.4.4.2 and the LOGIC SYSTEM FUNCTIONAL TEST performed in SR 3.3.6.4.4 overlap this surveillance to provide complete testing of the assumed safety function. If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 18 months on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 1). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Sections III and XI.

2. USAR, Chapter 15, Appendix 15B.

3. USAR, Section 15.

4. NRC Safety Evaluation to NEDC-31753P, March 8, 1993.

BASES

BACKGROUND (continued)

The drywell floor drain sump level monitoring system contains sump level instrumentation. The drywell floor drain sump level monitoring system provides two separate indications of LEAKAGE to the control room. First the level instrumentation provides input into a differentiation circuit that indicates the rate of change of the drywell floor drain sump level. This leakage rate is displayed in the control room. An alarm is provided if the leakage rate exceeds a preset limit. In addition, sump level is indicated in the control room. This sump can also be used to determine leak rates by determining how much the sump level has changed over any specific period of time, and thus establishing the leak rate associated with the level change. Either of these two automatic methods of quantifying leak rates are acceptable for determining the unidentified LEAKAGE in accordance with the requirements of LCO 3.4.5, and thus if either of these two automatic methods are available, the drywell floor drain sump monitoring system can be considered OPERABLE. If the drywell floor drain sump monitoring system is inoperable, manual methods may be utilized. Such manual methods may be necessary when sump level switches are out-of-service such that operator actions are necessary to determine in leakage (manually starting or stopping the sump pump, or manually timing its operating time). To employ such manual methods, the pumping rate of the pump must have been determined within the last fuel cycle (i.e., approximately 18 months).

The drywell atmospheric monitoring systems continuously monitor the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmospheric particulate and gaseous radioactivity monitoring systems are not capable of quantifying leakage rates, but are sensitive enough to indicate qualitative increases in LEAKAGE rates, on the order of 3 gpm within 1 hour.

(continued)

BASES

BACKGROUND (continued)

Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3). Condensate from the two upper drywell air coolers is routed to the drywell floor drain sump and is monitored by a flow transmitter that provides indications and alarms in the control room. This upper drywell air cooler condensate flow rate monitoring system serves as an added qualitative indicator, but not quantifier, of RCS unidentified LEAKAGE.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits. The systems either provide appropriate alarm of excess LEAKAGE in the control room, or they are monitored at appropriate intervals to identify excess LEAKAGE.

Identification of the LEAKAGE allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6).

Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, one of the two automatic methods of determining floor drain sump in leakage must be OPERABLE. The other monitoring systems provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.5. This Applicability is consistent with that for LCO 3.4.5.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, manual methods of determining the sump in leakage rate can provide the equivalent information to quantify leakage. In addition, the drywell atmospheric particulate or atmospheric gaseous monitor and the upper drywell air cooler condensate flow rate monitor will provide indications of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.5.1) using alternate methods such as the pump timer, operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still operable. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1

With both particulate and gaseous drywell atmospheric monitoring channels inoperable, grab samples of the drywell atmosphere shall be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed every 24 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., upper drywell air cooler condensate flow rate monitoring system) is available. The 24 hour interval provides periodic information that is adequate to detect LEAKAGE.

(continued)

BASES

ACTIONS

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

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as contributing to the alternate method capability. Alternate methods that can be used include (but are not limited to) the Reactor Water Cleanup System.

The plant is also required to enter Mode 4. This action is required because the alternate methods of decay heat removal may not be as reliable as the RHR shutdown cooling subsystems.

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

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or one recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the reactor coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

(continued)

BASES

BACKGROUND (continued)

The HPCS System (Ref. 3) consists of a single motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suction source to the sparger. Suction piping is provided from the CST and the suppression pool. Pump suction can be aligned to either the suppression pool or the CST. However, if the CST volume is low or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCS System. The HPCS System is designed to provide core cooling over a wide range of RPV pressures (0 psid to 1200 psid, vessel to suction source). Upon receipt of an initiation signal, the HPCS pump automatically starts after AC power is available and valves in the flow path begin to open. Since the HPCS System is designed to operate over the full range of expected RPV pressures, HPCS flow begins as soon as the necessary valves are open. Full flow test lines are provided to route water from and to the suppression pool or CST to allow testing of the HPCS System during normal operation without spraying water into the RPV.

The ECCS pumps are provided with minimum flow lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of 8 of the 19 S/RVs. It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (LPCS and LPCI), so that these subsystems can provide core cooling. Each ADS valve is supplied with pneumatic power from an air storage system, which consists of air accumulators located in the drywell.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during MODE 3 if necessary.

SR 3.5.1.3

Verification every 31 days that ADS accumulator supply pressure is ≥ 150 psig assures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The designed pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure (Ref. 13). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of 150 psig is provided by the Safety Related Instrument Air System. The 31 day Frequency takes into consideration administrative control over operation of the Safety Related Instrument Air System and alarms for low air pressure.

SR 3.5.1.4

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50, Appendix K, criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 10).

The ECCS pump differential pressure for each listed system in the Surveillance Requirements (SRs) 3.5.1.4 and 3.5.2.5, is the difference between the containment wetwell pressure and the RPV pressure assumed in the LOCA analyses at the time of injection/spray. In addition to this listed differential pressure, the ECCS pumps also need to overcome

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

elevation head loss and piping system friction loss at the required flow rate. This safety analysis value is determined by engineering calculation. In addition, pump operability may be limited by the ASME "required action" range value for these pumps. The Frequency for this Surveillance is in accordance with the Inservice Testing Program requirements.

SR 3.5.1.5

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance test verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCS, LPCS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup, and actuation of all automatic valves to their required positions. This Surveillance also ensures that the HPCS System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," overlaps this Surveillance to provide complete testing of the assumed safety function.

HPCS testing may be performed in any MODE. The Frequency of 18 months is based upon operating experience that has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

With the exception of the HPCS LOGIC SYSTEM FUNCTIONAL TEST, the 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 ans SR 3.5.2.2 (continued)

≥ 16 ft 6 inches or the HPCS System is aligned to take suction from the CST and the CST contains ≥ 220,000 gallons of water, assuring 150,000 gallons of water available for HPCS, equivalent to a volume of 47%, ensures that the HPCS System can supply makeup water to the RPV.

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool and CST water level variations during the applicable MODES.

Furthermore, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to an abnormal suppression pool or CST water level condition.

SR 3.5.2.3

The Bases provided for SR 3.5.1.1 is applicable to SR 3.5.2.3.

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper system response time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued)

performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve alignment would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

In MODES 4 and 5, the RHR System may operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, RHR valves that are required for LPCI subsystem operation may be aligned for decay heat removal. This SR is modified by a Note that allows one LPCI subsystem to be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. This will ensure adequate core cooling if an inadvertent vessel draindown should occur.

SR 3.5.2.5 and SR 3.5.2.6

The Bases provided for SR 3.5.1.4 and SR 3.5.1.5 are applicable to SR 3.5.2.5 and SR 3.5.2.6, respectively.

REFERENCES

1. USAR, Section 6.3.3.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the reactor vessel head spray nozzle. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST volume is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line A, upstream of the inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, 165 psia to 1215 psia. Rated flow is required up to 1118 psia, based on operation of the Safety Relief Valves in the Relief and Low-Low-Set modes (T.S. 3.3.6.4) during the vessel isolation transients for which RCIC is designed. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment–Operating

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Coolant System following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The primary containment consists of a free standing steel cylinder with an ellipsoidal dome, secured to a steel lined reinforced concrete mat, which surrounds the Reactor Coolant System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

The isolation devices for the penetrations in the primary containment boundary are a part of the primary containment leak tight barrier. To maintain this leak tight barrier:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks";

c. The equipment hatch is closed and sealed;

d. The leakage control systems associated with penetrations are OPERABLE, except as provided in LCO 3.6.1.8, "Feedwater Leakage Control System";

(continued)

BASES

BACKGROUND (continued)

e. The containment leakage rates are in compliance with the requirements of Specification 3.6.1.1 and Specification 3.6.1.3;

f. The suppression pool is OPERABLE; and

g. The sealing mechanism associated with each primary containment penetration, e.g., welds, bellows, or O-rings, is functional.

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a mechanistic fission product release following a DBA, based on NUREG 1465, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 0.20% by weight of the containment and drywell air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 7.80 psig (Ref. 4).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs, are a loss of coolant accident (LOCA), a main steam line break (MSLB), and a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside primary containment (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The inboard 42 inch purge supply and exhaust valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, and 3.

The outboard MSIVs must have a safety related air source available for use following an accident in order for leakage to be within limits. Therefore, anytime that this air source from the "B" train of P57 Safety Related Air System is not available, the outboard MSIVs may not be able to maintain valve leakage within the specified limits.

PCIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated

(continued)

BASES

LCO (continued)

The normally closed PCIVs or blind flanges are considered OPERABLE when, as applicable, manual valves are closed or opened in accordance with applicable administrative controls, automatic valves are de-activated and secured in their closed position, check valves with flow through the valve secured, or blind flanges are in place. The valves covered by this LCO with their associated stroke times, if applicable, are listed in Reference 3. Primary containment purge valves with resilient seals, secondary containment bypass valves, MSIVs, and hydrostatically tested valves must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment-Operating," as Type B or C testing.

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory, and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE and the primary containment purge valves are not required to be sealed closed in MODES 4 and 5. Certain valves are required to be OPERABLE, however, to prevent inadvertent reactor vessel draindown and release of radioactive material during a postulated fuel handling accident involving handling of recently irradiated fuel. These valves are those whose associated instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.) Due to radioactive decay, handling of fuel only requires containment isolation valve OPERABILITY when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) except for the inboard 42 inch (1M14-F045 and 1M14-F085) inch primary containment purge supply and exhaust isolation valve flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. Due to the size of the containment purge supply and exhaust

(continued)

BASES

ACTIONS (continued)

F.1, G.1, and G.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable, action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valves.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1

Each inboard 42 inch (1M14-F045 and 1M14-F085) primary containment purge supply and exhaust isolation valve is required to be verified sealed closed at 31 day intervals because the primary containment purge valves are not fully qualified to close under accident conditions. This SR is designed to ensure that a gross breach of primary containment is not caused by an inadvertent or spurious opening of a primary containment purge valve. Detailed analysis of these purge supply and exhaust isolation valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Primary containment purge valves that are sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The 31 day Frequency is based on primary containment purge valve use during unit operations.

This SR allows a valve that is open under administrative controls to not meet the SR during the time the valve is open. Opening a purge valve under administrative controls

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.1 (continued)

is restricted to one valve in a penetration flow path at a given time (refer to discussion for Note 1 of the ACTIONS) in order to effect repairs to that valve. This allows one purge valve to be opened without resulting in a failure of the Surveillance and resultant entry into the ACTIONS for this purge valve, provided the stated restrictions are met. Condition D must be entered during this allowance, and the valve opened only as necessary for effecting repairs. Each purge valve in the penetration flow path may be alternately opened, provided one remains sealed closed, if necessary, to complete repairs on the penetration.

The SR is modified by a Note stating that the inboard 42 inch primary containment purge supply and exhaust isolation valves are only required to be sealed closed in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves and the subsequent release of radioactive material will exceed limits prior to the closing of the purge valves. At other times when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies), pressurization concerns are not present and the purge valves are allowed to be open.

SR 3.6.1.3.2

This SR verifies that the 18 inch (1M14-F190, 1M14-F195, 1M14-F200, and 1M14-F205) and outboard 42 inch (1M14-F040 and 1M14-F090) primary containment purge supply and exhaust isolation valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have purge valve leakage outside the limits (Condition D).

The SR is also modified by a Note (Note 1) stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. At times other than MODE 1, 2, or 3 when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open (automatic isolation capability would be required by SR 3.6.1.3.5, SR 3.6.1.3.7, and SR 3.6.1.3.8).

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.3 (continued)

verified to be in the proper position, is low. A third Note is included to clarify that PCIVs open under administrative controls are not required to meet the SR during the time the PCIVs are open.

SR 3.6.1.3.4

This SR verifies that each primary containment isolation manual valve and blind flange located inside primary containment, drywell, or steam tunnel, and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For devices inside primary containment, drywell, or steam tunnel, the Frequency of "prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days," is appropriate since these devices are operated under administrative controls and the probability of their misalignment is low.

Four Notes are added to this SR. Note 1 provides an exception to meeting this SR in MODES other than MODES 1, 2, and 3. When not operating in MODES 1, 2, or 3, the primary containment boundary, including verification that required penetration flow paths are isolated, is addressed by LCO 3.6.1.10, "Primary Containment- Shutdown" (SR 3.6.1.10.1). The second Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in their proper position, is low. A third Note is included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open.

A fourth Note addresses removal of the Inclined Fuel Transfer System (IFTS) blind flange in MODES 1, 2, and 3. Requiring the Fuel Handling Building Fuel Transfer Pool water level to be ≥ 40' above the bottom of the pool ensures sufficient submergence of water over the bottom gate valve in the transfer tube to prevent direct communication between the Containment Building atmosphere and the Fuel

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.4 (continued)

Handling Building atmosphere, even upon occurrence of the peak post-accident pressure, P_a. Forty feet (40') above the bottom of the pool is equivalent to 22' 8 ¼" above the top of the flange for the IFTS bottom gate valve, which is approximately 3' 10" more water than needed to counteract the peak accident pressure of 7.8 psig. Also, since the IFTS drain piping does not have the same water seal as the transfer tube, administrative controls are required to ensure that the drain flow path can be quickly isolated whenever necessary.

These controls consist of designating an individual, whenever the 1F42-F003 valve is to be opened with the blind flange removed in MODE 1, 2, or 3, to be responsible for verifying closure of the valve if an accident occurs. This designated individual will remain in continuous communication with the control room, and be located at the 620' elevation in the Fuel Handling Area of the Intermediate Building. This person will be in addition to the minimum shift crew composition required to be at the plant site. Once the designated person is notified by the control room of the occurrence of an accident, his only assigned function will be to close this valve. The designated individual will verify the valve is closed from the controls at the IFTS panel if they are available. If this is not successful, the valve will be closed manually at the valve location. The designated person will be equipped with portable lighting (e.g., a flashlight) to supplement emergency lighting.

Also, the drain piping motor-operated isolation valve is tested in accordance with the Primary Containment Leak Rate Test Program. The leakage rate on this valve will be controlled by the strict limits on potential secondary containment bypass leakage (SR 3.6.1.3.9). Thus, the combination of water seal in the Fuel Handling Building, pressure integrity of the IFTS transfer tube, and administrative controls on the motor-operated drain valve in the drain piping, creates an acceptable barrier against post-accident leakage to the environment.

SR 3.6.1.3.5

Verifying the isolation time of each power operated and each automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.5 (continued)

full closure isolation time is demonstrated by SR 3.6.1.3.7. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.1.3.6

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened. Additionally, a leak rate acceptance criteria of 0.05 L_a has been assigned to these valves.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of recently irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.7

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

device. If both isolation devices in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two devices.

A Note is added to this SR which states that these valves are only required to meet this leakage rate limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary leakage rate limits are not required. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

A second Note makes it clear that Main Steam Line leakage need not be added into the secondary containment bypass leakage total, since Main Steam Line leakage is addressed separately in the radiological dose calculations; is not assumed to be immediately released to the environment like bypass leakage is; and because it is separately measured in SR 3.6.1.3.10.

SR 3.6.1.3.10

The analyses in References 1 and 2 are based on leakage that is less than the specified leakage rate. Leakage through each main steam line must be ≤ 100 scfh when tested at ≥ P_a, and the total leakage rate through all four main steam lines is ≤ 250 scfh. If the leakage rate on any Main Steam Line exceeds 100 scfh, the leakage rate will be restored to within 25 scfh when tested at ≥ P_a. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

The outboard MSIVs must have a safety related air source available for use following an accident in order for leakage to be within limits. Therefore, anytime that this air source from the "B" train of P57 Safety Related Air System is not available, the outboard MSIVs may not be able to meet this surveillance requirement.

A Note is added to this SR which states that these valve are only required to meet this leakage rate limit in MODES 1, 2, and 3. In other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage rate limits are not required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 is met. The combined leakage rate must be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

demonstrated at the frequency of the leakage test requirements of the Primary Containment Leakage Rate Testing Program.

This SR is modified by a Note that states these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage rate limits are not applicable in these other MODES or conditions.

A second Note states that the Feedwater lines are excluded from this particular hydrostatic (water) testing program. This is because water leakage from the stem, bonnet and seat of the third, high integrity valves in the feedwater lines (the gate valves) is controlled by the Primary Coolant Sources Outside Containment Program (Technical Specicification 5.5.2). The acceptance criteria for the Primary Coolant Sources Outside Containment Program is 7.5 gallons per hour.

SR 3.6.1.3.12

Verifying that each outboard 42 inch (1M14-F040 and 1M14-F090) primary containment purge supply and exhaust isolation valve is blocked to restrict opening to ≤ 50° is required to ensure that the valves can close under DBA conditions within the time limits assumed in the analyses of References 2 and 3.

The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when the purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies in the primary containment), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that the 2 inch Backup Hydrogen Purge System isolation valves are closed as required, or, if open, open for an allowable reason. These backup hydrogen purge isolation valves are fully qualified to close under accident conditions; therefore, these valves are allowed to be open for limited periods of time. This SR has been modified by a Note indicating the SR is not required to be met when the backup hydrogen purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or surveillances or special testing of the Backup Hydrogen Purge System (e.g., testing of the containment and drywell ventilation radiation monitors) that require the valves to be open. The 31 day Frequency is consistent with other drywell purge valve requirements.

REFERENCES

1. USAR, Chapter 15.
2. USAR, Section 6.2.
3. Plant Data Book, Tab G.
4. 10 CFR 50, Appendix J, Option B.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.7 Residual Heat Removal (RHR) Containment Spray System

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the primary containment airspace, bypassing the suppression pool. The primary containment also must withstand a low energy steam release into the primary containment airspace. The RHR Containment Spray System is designed to mitigate the effects of bypass leakage and low energy line breaks.

The RHR containment spray mode is operated post-LOCA, for up to 24 hours, in order to scrub released radionuclides from the containment atmosphere and into the suppression pool, and thus reduce the post-LOCA off-site and Control Room dose. Post-LOCA manual initiation for this function is based on a high radiation signal in the containment.

There are two redundant, 100% capacity RHR containment spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, two heat exchangers in series, and three spray spargers inside the primary containment (outside of the drywell). Dispersion of the spray water is accomplished by 346 nozzles in subsystem A and 344 nozzles in subsystem B.

The RHR containment spray mode will be automatically initiated for containment pressure reduction (based on pressure instrumentation), if required, following a LOCA. Containment spray is manually initiated for containment atmosphere post-LOCA dose mitigation, if required.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be 1.68 ft^2. The analysis demonstrates that

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

with containment spray operation the primary containment pressure remains within design limits.

The RHR Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential drywell bypass leakage paths and maintain the primary containment peak pressure below design limits, and provide for containment atmosphere dose reduction. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when the RHR pump, two heat exchangers in series, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

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

ACTIONS

A.1

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

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Feedwater Leakage Control System (FWLCS)

BASES

BACKGROUND

The FWLCS supplements the isolation function of the motor-operated primary containment isolation valves (PCIVs) in the feedwater lines that also penetrate the secondary containment. The motor-operated valve bonnets and internal seating volumes are sealed by water from the FWLCS to prevent fission products leaking past the isolation valves and bypassing the secondary containment after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The FWLCS consists of two independent, manually initiated subsystems, either of which is capable of preventing fission product leakage from the containment post LOCA. Each subsystem uses an ECCS water leg pump and a header which provides sealing water to pressurize the feedwater motor-operated valve bonnets and internal seating volumes.

APPLICABLE SAFETY ANALYSES

The analyses described in Reference 1 provide the evaluation of offsite dose consequences during accident conditions. For the Feedwater piping, a water seal would be maintained by the feedwater system outside the containment during the initial hour after a LOCA. That is, if the feedwater system becomes inoperable during the rapid vessel depressurization following a LOCA, the water within the feedwater piping will begin to flash into the drywell. A water seal would remain for a sufficient length of time followng the accident until the operator remotely isolates the motor-operated valve. Thus, a water seal would exist in the piping beyond the motor-operated valve. Initiation of the FWLCS then provides the water seal for the remainder of the 30 days of the accident. The offsite dose consequence calculations include consideration of any FWLCS water leakage past the seats of the gate valves.

The FWLCS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two FWLCS subsystems must be OPERABLE such that in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. A FWLCS subsystem is OPERABLE when all necessary components are available to supply each feedwater motor-operated valve with sufficient sealing

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Proper operation of the ECCS water leg pump is required to verify the capability of the FWLCS to provide sufficient sealing water to each feedwater motor-operated containment isolation valve to initiate and maintain the fluid seal for long term leakage control. The 31 day frequency is considered adequate based on operating experience, on the procedural controls governing ECCS operation, and on the low probability of major changes in the water leg pump capability during the period.

REFERENCES

1. USAR, Section 15.6.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.9 Main Steam Shutoff Valves

BASES

BACKGROUND

The post accident function of the Main Steam Shutoff Valves (MSSVs) is to be remote-manually closed in order to provide a reduction of post accident dose associated with the main steam line leakage path. With the Shutoff valves in a closed position, mitigation of the off-site and Control Room dose is achieved by taking credit for the deposition of particulate forms of released fission products (aerosols) on the inner walls of the four main steam lines. This removal process is based on a "plug flow" model for the Main Steam Isolation Valve (MSIV) leakage with relatively even, slow cooldown of the insulated main steam lines. The closed position of the MSSV supports the plug flow model by providing isolation of the space just upstream of the MSSV from external convection which could originate from the downstream nonsafety side of the MSSV. Therefore, the MSSVs are required to move to a closed position, but are not required or credited with any tightness against leakage.

The Main Steam Shutoff Valves (1N11-F0020 A, B, C, and D) are safety-class, remote manual motor-operated valves. The operator response to provide the manually initiated closure for the MSSVs is 20 minutes post-LOCA (Ref. 1). Failure of all four motor-operated MSSVs to close is taken as a single active failure based on a single operator error or on a loss of divisional power. This failure is not coincident with a single MSIV failure to close.

APPLICABLE SAFETY ANALYSES

The applicable safety analyses are the off-site and Control Room radiolgical dose calculations. The MSIVs in the main steam lines are required to close when a design basis accident (DBA) occurs. Failure to close an MSIV would

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

affect the retention of the fission product aerosols in that main steam line (and therefore the fission product release to the environment), unless a holdup volume could be established downstream of the closed MSIV in that line, i.e., using the Main Steam Shutoff Valve (MSSV). In determining the most limiting single failure case that would result in the highest (most conservative) calculated off-site doses, two cases were examined.

1. The single MSIV failure to close case results in less off-site and Control Room dose (than failure of all four MSSVs to close) because of credit for particulate deposition in the downstream volumes out to the closed MSSVs.

2. Failure of all four MSSVs to close is taken as a single active failure based on a single operator error or on a loss of divisional power, and results in the most limiting dose consequences. This failure is not coincident with a single MSIV failure to close. This limiting case assumes that main steam line leakage is attenuated in the main steam line from the reactor vessel out to the outboard MSIV. Although this most limiting analysis case assumed a failure to close the MSSVs, retention of OPERABILITY requirements on these valves is appropriate to ensure the single failure analysis associated with the LOCA off-site and Control Room dose reanalysis remains valid. The Main Steam Shutoff Valves meet Criterion 3 of 10CFR50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The four MSSVs are part of the mitigation strategy for off-site and Control Room dose consequences. Failure of all four MSSVs to close is taken as a single active failure based on a single operator error or on a loss of divisional power, and results in the most limiting dose consequences. Although this most limiting analysis case assumed a failure to close the MSSVs, retention of OPERABILITY requirements on these valves is appropriate to ensure the single failure analysis associated with the LOCA off-site and Control Room dose reanalysis remains valid.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSSV OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSSVs OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.

ACTIONS

The ACTIONS are modified by a Note allowing separate condition entry for each penetration flow path because an inoperable Main Steam Shutoff Valve (MSSV) in a main steam line does not affect the ability to provide a post-accident holdup volume in the affected line or in the other lines (between the MSIVs or between an MSIV and an MSSV). The Required Actions provide appropriate compensatory actions for each inoperable MSSV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable MSSVs are governed by subsequent Condition entry and application of associated Required Actions.

A.1

Each Main Steam Line has two Main Steam Isolation Valves (MSIVs) and a downstream Main Steam Shutoff Valve (MSSV), i.e., 1N11-F0020 A, B, C, or D. With one or more MSSVs inoperable, the inoperable MSSV must be restored to OPERABLE status or the Main Steam Line must be isolated within 30 days. During this 30 day Completion Time, the remaining OPERABLE MSIVs in that Main Steam Line are adequate to

(continued)

BASES

ACTIONS (continued)

perform the required leakage holdup function should a LOCA occur. However, the overall reliability is reduced because a single failure of an MSIV in that line could result in a perform the required leakage holdup function should a LOCA loss of the MSIV leakage holdup function, because post-accident, a single MSIV failure would prevent establishment of a "holdup volume". If an MSSV is "inoperable", the action of closing the MSSV in that main steam line is based on the characteristics of the revised design basis accident source term (i.e., predominantly aerosol). Closing the MSSV will provide for a post-LOCA single-failure-proof holdup volume within the main steam line, for deposition of the aerosol on the inner walls of the main steam line. The Required Action does not permit closure of an MSSV and an MSIV at the same time, or both MSIVs at the same time, since doing so during plant operation could result in differential cooldown of that particular Main Steam Line, with resultant damage to some small-bore drain piping that is interconnected to the other Main Steam Lines. If an MSSV is "inoperable", but closed, credit can be taken for it in meeting the ACTION. Leak tightness of the MSSVs is not necessary to ensure the assumptions of the dose calculation methodology are met for the main steam lines, since leakage flow characteristics used in the analyses are affected only by the turbulence caused by an open ended pipe (i.e., the Main Steam Shutoff Valves fail to close).

The 30 day Completion Time is based on the redundant capability afforded by the OPERABLE MSIVs on that line, and the low probability of a DBA LOCA occurring during this period. After 30 days, when the MSSV on that line has been closed, a post-LOCA holdup volume can be established without concern over a single failure, therefore plant operation may continue.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1

The only necessary surveillance requirement is one to ensure the Main Steam Shutoff Valves will stroke closed on a manual demand by the operators. Leak test requirements are not necessary to ensure the assumptions of the dose calculation methodology are met for the main steam lines, since leakage flow characteristics used in the analyses are affected only by the turbulence caused by an open ended pipe (i.e., the Main Steam Shutoff Valves fail to close). The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES (continued)

REFERENCES

1. Calculation PSAT 04202H.08 "Steamline: Particulate Decontamination Control"

2. Calculation PSAT 04202H.13 "Offsite and Control Room Dose Calculation"

3. Calculation PSAT 08401T.03 "Perry Plant Total Effective Dose Equivalent (TEDE) Calculation"

4. USAR Section 15.6.5

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.10 Primary Containment–Shutdown

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Coolant System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material to within limits. The primary containment surrounds the Reactor Coolant System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions.

The isolation devices for the penetrations in the primary containment boundary are a part of the primary containment leak tight barrier. To maintain this leak tight barrier for accidents during shutdown conditions:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks"; and

c. The equipment hatch is closed.

(continued)

BASES

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a fuel handling accident involving handling of recently irradiated fuel, or reactor vessel draindown, provides an acceptable leakage barrier to contain fission products, thereby minimizing offsite doses.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it contain the fission products from a fuel handling accident involving handling of recently irradiated fuel inside the primary containment (Ref. 2), to limit doses at the site boundary to within limits. The primary containment OPERABILITY in conjunction with the automatic closure of selected OPERABLE containment isolation valves (LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"), assures a leak tight fission product barrier. Its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage rates assumed in safety analyses.

The fuel handling accident inside the primary containment has been analyzed for two cases. In the first scenario, the fuel bundles involved are recently irradiated, i.e., they have occupied part of a critical reactor core within the previous seven days. The containment purge system is in operation and isolates on high radiation. This produces an immediate unfiltered release to the environment. The fission products which remain within the primary containment are conservatively assumed to be released at rates consistent with the DBA LOCA assumptions (e.g., 0.2% of the containment volume per day), and be filtered by the Annulus Exhaust Gas Treatment System prior to release to the environment.

In the second case, the fuel handling accident inside the primary containment is assumed to involve fuel bundles that have not been in a critical reactor core within the previous seven days. With the radioactive decay provided with this delay, all gaseous fission products released from the damaged fuel bundles are assumed to be immediately discharged directly to the environment.

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Primary containment OPERABILITY is maintained by providing a contained volume to limit fission product escape following a fuel handling accident involving handling of recently irradiated fuel, or an unanticipated water level excursion. Compliance with this LCO will ensure a primary containment configuration, including the equipment

(continued)

BASES

LCO (continued)

hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since offsite dose analyses conservatively assume LOCA leakage pathways and rates, the isolation and closure times of automatic containment isolation valves supports an OPERABLE primary containment during shutdown conditions. Furthermore, normal operation of the inclined fuel transfer system (IFTS) without the IFTS blind flange installed is considered acceptable for meeting Primary Containment-Shutdown OPERABILITY.

Leakage rates specified for the primary containment and air locks, addressed in LCO 3.6.1.1 and LCO 3.6.1.2 are not directly applicable during the shutdown conditions addressed in this LCO.

APPLICABILITY

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of Primary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the primary containment is not required to be OPERABLE.

ACTIONS

A.1 and A.2

In the event that primary containment is inoperable, action is required to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.10.1

This SR verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed manual valve, a closed and de-activated automatic valve, and a blind flange. This SR does not require any testing or isolation device manipulation. Rather, it involves verification that these isolation devices capable of being mispositioned are in the correct position. The 31 day Frequency was chosen to provide added assurance that the isolation devices remain in the correct positions.

This SR is modified by three Notes. The first Note does not require this SR to be met for pathways capable of being isolated by OPERABLE primary containment automatic isolation valves. The second Note permits the Fire Protection System manual hose reel containment isolation valves (1P54-F726 and 1P54-F727) to be open during shutdown conditions to supply fire mains. The third Note is included to clarify that manual valves opened under administrative controls are not required to meet the SR during the time the manual valves are open.

REFERENCES

1. Deleted.

2. USAR, Section 15.7.6.

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Inadvertent actuation of both primary RHR containment spray subsystems during normal operation;

The results of these two cases show that the containment vacuum breakers, with an opening setpoint of 0.1 psid, are capable of maintaining the differential pressure within design limits.

The containment vacuum breakers satisfy Criterion 3 of the NRC Policy Statement.

LCO

Only 3 of the 4 vacuum breakers must be OPERABLE for opening. All containment vacuum breakers, however, are required to be closed (except during testing or when the vacuum breakers are performing their intended design function). The vacuum breaker OPERABILITY requirement provides assurance that the containment negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

APPLICABILITY

In MODES 1, 2, and 3, the RHR Containment Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the containment could occur due to inadvertent actuation of this system. The vacuum breakers, therefore, are required to be OPERABLE in MODES 1, 2, and 3, to mitigate the effects of inadvertent actuation of the RHR Containment Spray System.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining containment vacuum breakers OPERABLE is not required in MODE 4 or 5.

When handling recently irradiated fuel in the primary containment, and during operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Containment vacuum breakers are therefore required to be OPERABLE during these evolutions to protect the primary containment against an inadvertent initiation of the Containment Spray System. Due to radioactive decay, handling of fuel only requires OPERABILITY of Containment Vacuum Breakers when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. Since OPDRVs assume that one or more fuel assemblies are loaded into the core, this LCO would not be applicable for OPDRVs if no fuel is in the reactor vessel.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Note has been added to provide clarification that separate Condition entry is allowed for each containment vacuum breaker.

B.1 and B.2

If the Required Action of Condition A cannot be met, or if there are three or more containment vacuum breakers not closed, or if there are two or three required vacuum breakers inoperable for other reasons, the plant must be brought to a MODE or condition in which the LCO does not apply. To achieve this status, if the plant is operating, ACTION B.1 requires that the plant be brought to at least MODE 3 within 12 hours and that the plant be brought to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. A Note has been added to stipulate that these Required Actions are only applicable if the plant is in MODE 1, 2, or 3.

If the Condition occurs during movement of recently irradiated fuel in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs), then ACTION B.2 requires that action be taken to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel in the primary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be taken to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended. A Note has been added to the Required Actions to stipulate that these requirements are only applicable while moving recently irradiated fuel assemblies in the primary containment, or during OPDRVs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.11.3 (continued)

open at a differential pressure of ≤ 0.2 psid (outside containment to containment) is valid. Verification that the vacuum breaker isolation valves will open assures that the vacuum breakers are available to perform their intended function. Two of the vacuum breaker isolation valves have an opening allowable value of ≥0.052 psid and ≤0.148 psid, while the other two vacuum breaker isolation valves have an opening allowable of ≥0.064 psid and ≤0.160 psid (containment to outside containment).

Performance of this SR includes a CHANNEL CALIBRATION of the isolation valve actuation instrumentation. 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 at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 6.2.1.1.4.2.

BASES

LCO (continued)

relative humidity are not required to be maintained within the prescribed limits.

APPLICABILITY

In MODES 1, 2, and 3, the RHR Containment Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the containment could occur due to inadvertent actuation of this system. The containment average temperature relationship with relative humidity, therefore, is required to be within limits in MODES 1, 2, and 3, to mitigate the effects of inadvertent actuation of the RHR Containment Spray System.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES. Therefore, maintaining limits on containment relative humidity and temperature is not required in MODE 4 or 5.

When handling recently irradiated fuel in the primary containment, and during operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Therefore, the proper relationship between containment average temperature and relative humidity must exist during these evolutions. Due to radioactive decay, handling of fuel only requires control over Containment humidity when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

ACTIONS

A.1

With the primary containment average temperature and relative humidity not within the established limits, actions must be taken to restore the primary containment relative humidity and temperature to within limits. With the plant operating in MODE 1, 2, or 3, Required Action A.1 stipulates that restoration must occur within 8 hours. The eight hour Completion Time is based on the time required to restore the relative humidity and temperature limits, and the low probability of an event occurring during this time period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the primary containment relative humidity and temperature cannot be restored to within limits within the required Completion Time of Condition A, actions must be taken to place the plant in a MODE or condition in which the LCO does not apply.

Required Action B.1 requires that the plant be brought to at least MODE 3 within 12 hours and Required Action B.2 requires that the plant be brought to MODE 4 within 36 hours.

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

C.1 and C.2

If the primary containment relative humidity and temperature are not within limits during movement of recently irradiated fuel in the primary containment, or during OPDRVs, action is required to place the plant in a MODE or condition in which the LCO does not apply.

Required Actions C.1 and C.2 require that actions be taken to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel in the primary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be taken to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1 (continued)

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.1.2

This SR ensures that there are no physical problems (e.g., loose wiring or structural connections, deposits of foreign materials, etc.) that could affect primary containment hydrogen recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.1.3

This SR requires performance of a resistance to ground test of each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms following the performance of SR 3.6.3.1.1.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision 2.
4. USAR, Section 6.2.5.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.2.3 and SR 3.6.3.2.4

These functional tests are performed every 18 months to verify system OPERABILITY. The current draw to develop a surface temperature of ≥ 1700°F is verified for hydrogen igniters in inaccessible areas, e.g., in a high radiation area. Additionally, the surface temperature of each accessible hydrogen igniter is measured to be ≥ 1700°F to demonstrate that a temperature sufficient for ignition is achieved. 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 that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. USAR, Section 6.2.8.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.3 Combustible Gas Mixing System

BASES

BACKGROUND

The Combustible Gas Mixing System ensures a uniformly mixed post accident containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration.

The Combustible Gas Mixing System is an Engineered Safety Feature and is designed to operate following a loss of coolant accident (LOCA) in post accident environments without loss of function. There are two redundant and independent combustible gas mixing subsystems, each consisting of a compressor and associated valves, controls, and piping. Each combustible gas mixing subsystem is sized to pump 500 scfm. Each subsystem is powered from a separate emergency power supply. Since each combustible gas mixing subsystem can provide 100% of the mixing requirements, the system will provide its design function with a worst case single active failure.

Following a LOCA, the drywell is immediately pressurized due to the release of steam into the drywell environment. This pressure is relieved by the lowering of the water level within the weir wall, clearing the horizontal vents and allowing the mixture of steam and noncondensibles to flow into the primary containment through the suppression pool, removing much of the heat from the steam. The remaining steam in the drywell begins to condense. As steam flow from the reactor pressure vessel ceases, the drywell pressure falls rapidly. The combustible gas mixing compressors are manually started prior to the drywell hydrogen concentration exceeding 3.0 v/o. The compressors force air from the primary containment into the drywell. Drywell pressure increases until the water level between the weir wall and the drywell is forced down to the horizontal pool vents forcing drywell atmosphere back into containment and mixing with containment atmosphere to dilute the hydrogen. While combustible gas mixing continues following the LOCA, hydrogen continues to be produced. Eventually, the 4.0 v/o limit is again approached and the primary containment hydrogen recombiners are manually placed in operation.

(continued)

BASES

BACKGROUND (continued)

The containment spray is credited with removal of airborne radionuclides. Post-LOCA operation of the mixing compressors also provides a transport of air between containment and the drywell. Therefore, post-LOCA dose is reduced with mixing compressor operation.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Annulus Exhaust Gas Treatment (AEGT) System and manual closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs).

The secondary containment is a structure that completely encloses the primary containment. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the external pressure. To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Annulus Exhaust Gas Treatment (AEGT) System."

The isolation devices for the penetrations in the secondary containment boundary are a part of the secondary containment barrier. To maintain this barrier:

a. All penetrations terminating in the secondary containment required to be closed during accident conditions are closed by at least one manual valve or blind flange, as applicable, secured in its closed position, except as provided in LCO 3.6.4.2, Secondary Containment Isolation Valves (SCIVs)";

(continued)

BASES

BACKGROUND (continued)

b. The containment equipment hatch is closed and sealed and the shield blocks are installed adjacent to the shield building;

c. The door in each access to the secondary containment is closed, except for entry and exit;

d. The sealing mechanism associated with each shield building penetration, e.g. welds, bellows, or O-rings, is functional;

e. The pressure within the secondary containment is less than or equal to the value required by Surveillance Requirement SR 3.6.4.1.1, except for entry and exit to the annulus; and

f. The Annulus Exhaust Gas Treatment System is OPERABLE.

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1) and a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside primary containment (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the AEGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

(continued)

BASES

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPRDVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of Secondary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the secondary containment is not required to be OPERABLE.

ACTIONS

A.1

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

B.1 and B.2

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

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Movement of recently irradiated fuel assemblies in the primary containment and OPDRVs can be postulated to cause significant fission product releases. In such cases, the secondary containment is one of the barriers to release of fission products to the environment. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended if the secondary containment is inoperable. Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

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

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

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that the primary containment equipment hatch is closed and the shield blocks are installed adjacent to the shield building, and secondary containment access doors are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. In this application, the term "sealed" has no connotation of leak tightness. Verifying that all such openings are closed provides adequate

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1).

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Isolation barrier(s) for the penetration are discussed in Reference 2. The isolation devices addressed by this LCO are passive. Manual valves and blind flanges are considered passive devices.

Penetrations are isolated by the use of manual valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1), and a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) inside primary containment (Ref. 3). The secondary containment performs no active function in response to each of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Annulus Exhaust Gas Treatment (AEGT) System before being released to the environment.

Maintaining SCIVs OPERABLE ensures that fission products will remain trapped inside secondary containment so that they can be treated by the AEGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of the NRC Policy Statement.

(continue)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed, or open in accordance with appropriate administrative controls, or blind flanges are in place. The valves covered by this LCO are included in Table B 3.6.4.2-1.

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of secondary containment isolation valves when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the SCIVs are not required to be OPERABLE.

ACTIONS

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

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

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

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

C.1 and C.2

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

D.1 and D.2

If any Required Action and associated Completion Time of Condition A or B cannot be met during movement of recently irradiated fuel assemblies in the primary containment,

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

or during OPDRVs, the plant must be placed in a condition in which the LCO does not apply. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

Since these isolation devices are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the isolation devices are in the correct positions.

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

(continued)

BASES

BACKGROUND (continued)

humidity of the airstream to less than 70% (Ref. 2). The roughing filter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The AEGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. AEGT System flows are controlled by two motor operated control dampers installed in branch ducts. One duct exhausts air to the unit vent, (AEGT Subsystem A exhausts to the Unit 1 plant vent; AEGT Subsystem B exhausts to the Unit 2 plant vent), while the other recirculates air back to the annulus.

APPLICABLE SAFETY ANALYSES

The design basis for the AEGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days (Ref. 2). For all events analyzed, the AEGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The AEGT System satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the AEGT System OPERABLE is not required in MODE 4 or 5, except for

(continued)

BASES

APPLICABILITY (continued)

other situations under which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of the AEGT System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the AEGT System is not required to be OPERABLE.

ACTIONS

A.1

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

B.1 and B.2

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

C.1, C.2.1 and C.2.2

During movement of recently irradiated fuel assemblies in the primary containment, or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE AEGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no

(continued)

BASES

ACTIONS

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

failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material, thus placing the unit in a Condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies in the primary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

D.1

If both AEGT subsystems are inoperable in MODE 1, 2, or 3, the AEGT System may not be capable of supporting the required radioactivity release control function. Therefore, LCO 3.0.3 must be entered immediately.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each AEGT subsystem from the control room for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CRER System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters 6 and 15 (Refs. 3 and 4, respectively). The emergency recirculation mode of the CRER System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of ability to recirculate air in the control room.

The CRER System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant subsystems of the CRER System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in a failure to meet the dose requirements of GDC 19 in the event of a DBA.

The CRER System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A CRER subsystem is considered OPERABLE when its associated:

a. Fans are OPERABLE;

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

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

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, and 3, the CRER System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

(continued)

BASES

APPLICABILITY (continued)

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRER System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building; and

b. During operations with a potential for draining the reactor vessel (OPDRVs).

Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room Emergency Recirculation System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the CRER System is not required to be OPERABLE.

ACTIONS

A.1

With one CRER subsystem inoperable, the inoperable CRER subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CRER subsystem is adequate to perform control room radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE CRER subsystem could result in loss of CRER System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining CRER subsystem can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRER subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

C.1, C.2.1 and C.2.2

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or during OPDRVs, if the inoperable CRER subsystem cannot be restored to OPERABLE status within the required Completion Time of Condition A, the OPERABLE CRER subsystem may be placed in the emergency recirculation mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both CRER subsystems are inoperable in MODE 1, 2, or 3, the CRER System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or during OPDRVs, with two CRER subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Operating each CRER subsystem for ≥ 10 continuous hours after initiating from the control room and ensuring flow through the HEPA filters and charcoal adsorbers ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.7.3.2

This SR verifies that the required CRER testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRER filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the

(continued)

BASES (continued)

LCO

Two independent and redundant subsystems of the Control Room HVAC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room HVAC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the cooling coils, fans, chillers with compressors, ductwork, dampers, and associated instrumentation and controls. The heating coils are not required for control room HVAC OPERABILITY.

APPLICABILITY

In MODE 1, 2, or 3, the Control Room HVAC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room HVAC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building; and

b. During operations with a potential for draining the reactor vessel (OPRDVs).

Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room HVAC System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the Control Room HVAC System is not required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1

With one control room HVAC subsystem inoperable, the inoperable control room HVAC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room HVAC subsystem is adequate to perform the control room air

(continued)

BASES

ACTIONS (continued)

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

The Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or during OPDRVs, if the inoperable control room HVAC subsystem cannot be restored to OPERABLE status within the required Completion Time of Condition A, the OPERABLE control room HVAC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent 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 present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of recently irradiated fuel assemblies in the primary containment or fuel handling building, or during OPDRVs if the Required Action and associated Completion Time of Condition B is not met, action must be taken to immediately suspend activities that present a potential for releasing significant amounts of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, handling of recently irradiated fuel in the primary containment or fuel handling building must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analysis. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the Control Room HVAC System is not expected over this time period.

REFERENCES

1. USAR, Section 6.4.

2. USAR, Section 9.4.1.

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas release rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the USAR, Section 15.7.1 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The release rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits (NUREG-0800, Ref. 2) of 10 CFR 100 (Ref. 3), or the NRC staff approved licensing basis.

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement.

The Offgas limit specified in TS 3.7.5 represents a short term conservative limit for accident analysis purposes. The operational limits defined by TS sections 5.5.1 and 5.5.4 and by the ODCM ensure that the annual average Offgas release rates are significantly under this, and ensure consistency with the design bases for annual average release limits and shielding analyses.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission

(continued)

BASES

LCO (continued)

product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. The LCO is conservatively established at (3579 MWT x 100 μCi/MWt-second = 358 mCi/second).

APPLICABILITY

The LCO is applicable when steam is being exhausted to the main condenser and the resulting noncondensibles are being processed via the Main Condenser Offgas System. This occurs during MODE 1, and during MODES 2 and 3 with any main steam line not isolated and the SJAE in operation. In MODES 4 and 5, steam is not being exhausted to the main condenser and the requirements are not applicable.

ACTIONS

A.1

If the offgas radioactivity rate limit is exceeded, 72 hours is allowed to restore the release rate to within the limit. The 72 hour Completion Time is reasonable, based on engineering judgment considering the time required to complete the Required Action, the large margins associated with permissible dose and exposure limits, and the low probability of a Main Condenser Offgas System rupture occurring.

B.1, B.2, B.3.1, and B.3.2

If the release rate is not restored to within the limits within the associated Completion Time, all main steam lines or the SJAE must be isolated. This isolates the Main Condenser Offgas System from the source of the radioactive steam. The main steam lines are considered isolated if at least one main steam isolation valve in each main steam line is closed, and at least one main steam line drain valve in each drain line is closed. The 12 hour Completion Time is reasonable, based on operating experience, to perform the actions from full power conditions in an orderly manner and without challenging unit systems.

An alternative to Required Actions B.1 and B.2 is to place the unit in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

BACKGROUND (continued)

With the boundaries in place, the FHB Ventilation Exhaust System will assure that any releases occurring as a result of a FHA are filtered.

APPLICABLE SAFETY ANALYSES

There is only one principal accident for which credit is taken for FHB OPERABILITY. This is the FHA involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous seven days) in the FHB (Ref. 1). The FHB performs no active function in response to the FHA; however, proper air flow patterns are required to ensure that the release of radioactive materials is restricted to those leakage rates assumed in the accident analysis.

FHB satisfies Criterion 3 of the NRC Policy Statement.

LCO

An OPERABLE FHB provides a control volume into which fission products can be diluted and processed prior to release to the environment. For the FHB to be considered OPERABLE, it must provide proper air flow patterns to ensure that there is no uncontrolled release of radioactive material during a FHA involving handling of recently irradiated fuel in the FHB.

APPLICABILITY

In plant operating MODES, OPERABILITY of the FHB is not required since leakage from the primary containment will not be released into the FHB. Regardless of the plant operating MODE, anytime recently irradiated fuel is being handled in the FHB there is the potential for significant radioactive releases due to a FHA, and the FHB is required to mitigate the consequences.

Due to radioactive decay, handling of fuel only requires OPERABILITY of the Fuel Handling Building when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

(continued)

BASES (continued)

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

A.1

With the FHB inoperable, the plant must be brought to a condition in which the LCO does not apply since the FHB is incapable of performing its required accident mitigation function. To achieve this, handling of recently irradiated fuel must be suspended immediately. Suspension shall not preclude completion of fuel movement to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 and SR 3.7.8.2

Verifying that FHB floor hatches and access doors are closed, that the shield blocks are in place adjacent to the shield building, and that the FHB railroad track door is closed ensures that proper air flow patterns will exist in the FHB, and that any release following a FHA involving handling of recently irradiated fuel in the FHB will be filtered prior to release. Verifying that all such openings are closed provides adequate assurance that exfiltration from the FHB will not occur. Maintaining FHB OPERABILITY requires verifying each door in the access opening is closed, except when the access opening is being used for entry and exit.

The 24 hour Frequency for these SRs has been shown to be adequate based on operating experience.

REFERENCES

1. USAR, Section 15.7.4.

BASES

BACKGROUND (continued)	radiation condition, an alarm will occur in the control room, and the operating supply fan from the FHB Ventilation Supply System will trip. The exhaust subsystems remain operational to continue exhausting contaminated air from the fuel handling area through the charcoal filter trains, thus precluding any uncontrolled release of radioactivity to the outside environment.
APPLICABLE SAFETY ANALYSES	The design basis for the FHB Ventilation Exhaust System is to mitigate the consequences of a FHA involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days (Ref. 3). For all events analyzed, the FHB Ventilation Exhaust System reduces, via filtration and adsorption, the radioactive material released to the environment. The FHB Ventilation Exhaust System satisfies Criterion 3 of the NRC Policy Statement.
LCO	Following a FHA involving handling of recently irradiated fuel, a minimum of two FHB ventilation exhaust subsystems are required to maintain the FHB at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for three OPERABLE subsystems ensures operation of at least two FHB ventilation exhaust subsystems in the event of a single active failure.
APPLICABILITY	In plant operating MODES, OPERABILITY of the FHB Ventilation Exhaust System is not required since leakage from the primary containment will not be released into the FHB. Regardless of the plant operating MODE, anytime recently irradiated fuel is being handled in the FHB there is the potential for significant radioactive releases due to a FHA, and the FHB Ventilation Exhaust System is required to mitigate the consequences. Due to radioactive decay, handling of fuel only requires OPERABILITY of the Fuel Handling Building Ventilation Exhaust System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

(continued)

BASES(continued)

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

A.1

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

B.1 and B.2

If the FHB ventilation exhaust subsystem cannot be restored to OPERABLE status within the required Completion Time the two remaining OPERABLE FHB ventilation exhaust subsystems should be immediately placed in operation. This Required Action ensures that the remaining subsystems are OPERABLE, and that any other failure would be readily detected.

An alternative to Required Action B.1 is to immediately suspend activities that represent a potential for releasing significant amounts of radioactive material to the FHB, thus placing the unit in a condition that minimizes risk by suspending movement of recently irradiated fuel assemblies. Suspension of this activity shall not preclude completion of fuel movement to a safe position.

C.1

With two or three FHB ventilation exhaust subsystems inoperable the plant must be brought to a condition in which the LCO does not apply since the system is incapable of performing its required accident mitigation function. To achieve this, handling of recently irradiated fuel in the FHB must be suspended immediately. Suspension shall not preclude completion of fuel movement to a safe position.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The ECCW System satisfies Criterion 3 of the NRC Policy Statement.
LCO	In the event of a DBA, one ECCW subsystem 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 ECCW subsystems must be OPERABLE. At least one ECCW subsystem will operate assuming the worst single active failure occurs coincident with the loss of offsite power. An ECCW subsystem is considered OPERABLE when: a. The associated pump and surge tank are OPERABLE; and b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE. The isolation of ECCW to other components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the ECCW System.
APPLICABILITY	In MODE 1, the ECCW subsystems are in standby except when required to support RHR, LPCS, or RCIC System operations and testing. In MODES 2 and 3, the ECCW System is operated as necessary to support hot standby conditions or normal plant shutdown and cooldown using the RHR System. In MODES 4 and 5, the requirements of the ECCW System are determined by the systems they support (Ref. 2).

(continued)

BASES (continued)

ACTIONS

A.1

With one or both ECCW subsystems inoperable, all the associated subsystem(s) or component(s) must immediately be declared inoperable.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Verifying the correct alignment for each manual, power operated, and automatic valve in the ECCW subsystem flow path provides assurance that the proper flow paths exist for ECCW subsystem operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a non-accident position provided the valve will automatically reposition in the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

Isolation of the ECCW subsystem to components or systems does not necessarily affect the OPERABILITY of the ECCW subsystem. As such, when the ECCW subsystem pump, valves, and piping are OPERABLE, but a branch connection off the main header is isolated, the associated ECCW subsystem needs to be evaluated to determine if it is still OPERABLE.

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

SR 3.7.10.2

This SR verifies that the automatic isolation valves of the Division 1 and 2 ECCW subsystems will automatically realign to the safety or emergency position to provide cooling water exclusively to the safety related equipment during an accident. This is demonstrated by use of an actual or simulated initiation signal. This Surveillance also verifies the automatic start capability of the ECCW pump in each subsystem. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.5.1.6 overlaps this Surveillance to provide complete testing of the safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 9.2.2.
2. Plant Data Book, Tab R, Section 6.4.9.

BASES

BACKGROUND (continued)

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a LOCA.

Certain required plant loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System or to prevent overloading the DG.

Ratings for DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating is 7000 Kw for Divisions 1 and 2 and is 2600 Kw for Division 3, with 10% overload permissible for up to 2 hours in any 24 hour period.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the USAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit as discussed in Reference 2. This includes maintaining the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

AC sources satisfy the requirements of Criterion 3 of the NRC Policy Statement.

(continued)

BASES

ACTIONS (continued)

A.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. This Completion Time assumes sufficient offsite power remains to power the minimum loads needed to respond to analyzed events. In the event one or more divisions are without offsite power, this assumption is not met. Therefore, the optional Completion Time is specified. Should two (or more) divisions be affected, the 24 hour Completion Time is conservative with respect to the Regulatory Guide assumptions supporting a 24 hour Completion Time for both offsite circuits inoperable. 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.

The Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The third 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 contiguous 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 14 days. This situation could lead to a total of 17 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 14 days (for a total of 31 days) allowed prior to complete restoration of the LCO. The 17 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 17 day Completion Times for Required Action A.2 means that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either Required Actions B.3.1 or B.3.2, the corrective actions program will continue to evaluate the common cause failure 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. 7), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. The 72 hour and 14 day Completion Times take into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

The third 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 restored OPERABLE, the LCO may already have been not met for up to 72 hours. This situation could lead to a total of 17 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 20 days) allowed prior to complete restoration of the LCO. The 17 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 Completion Times means that the three Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

SR 3.8.1.7 requires that, at a 184 day Frequency, the Division 1 and 2 DGs start from standby conditions and achieves required voltage and frequency within 10 seconds. Also, this SR requires that the Division 3 DG starts from standby conditions and achieves a minimum required frequency within 10 seconds and required voltage and frequency within 13 seconds. The start time requirements support the assumptions in the design basis LOCA analysis (Ref. 5). The start time requirements are not applicable to SR 3.8.1.2 (see Note 3 of SR 3.8.1.2). Since SR 3.8.1.7 does require timed starts, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This procedure is the intent of Note 1 of SR 3.8.1.2. Similarly, the performance of SR 3.8.1.12 or SR 3.8.1.19 also satisfies the requirements of SR 3.8.1.2 and SR 3.8.1.7.

The 31 day Frequency for SR 3.8.1.2 is consistent with the industry guidelines for assessment of diesel generator performance (Ref. 14). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance demonstrates that the DGs are capable of synchronizing and accepting greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while 1.0 is an operational limitation to ensure circulating currents are minimized. The load band for the Division 1 and 2 DGs is provided to avoid routine overloading of these DGs. While this Surveillance allows operation of the Division 1 and 2 DGs in the band of 5600 kW to 7000 kW, a range of 5600 kW to 5800 kW will normally be used in order to minimize wear on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

the DGs. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with the industry guidelines for assessment of diesel generator performance (Ref. 14).

Note 1 modifies this Surveillance to indicate that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized.

Note 2 modifies this Surveillance by stating that momentary transients because of changing bus loads do not invalidate this test.

Note 3 indicates that this Surveillance shall 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 4 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 that will ensure transfer pump operation and availability. This level is expressed as an equivalent volume in gallons, and will ensure fuel oil transfer pump suction requirements are satisfied for all pump operating transients, including normal tank draw down during a secondary pump start.

The 31 day frequency is adequate to ensure that a sufficient supply of fuel oil is available. Subsequent to receipt of a diesel generator auto-initiation alarm, plant operators would be able to verify proper primary and secondary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

The reason for Note 2 is that credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds for Divisions 1 and 2 and 13 seconds for Division 3. The times are derived from the requirements of the accident analysis to respond to a design basis large break LOCA.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR has been modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The requirement that the diesel has operated for at least 1 hour at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. The load band for the Division 1 and 2 DGs is provided to avoid routine overloading of these DGs. While this Surveillance allows operation of the Division 1 and 2 DGs in the band of 5600 kW to 7000 kW, a range of 5600 kW to 5800 kW will normally be used in order to minimize wear on the DGs. This is the load range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.20 (continued)

continuously circulated and temperature maintained consistent with manufacturer recommendations for Division 1 and 2 DGs. For the Division 3 DG, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. USAR, Chapter 8.
3. Regulatory Guide 1.9.
4. USAR, Chapter 6.
5. USAR, Chapter 15.
6. Regulatory Guide 1.93.
7. Generic Letter 84-15, July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108.
10. Regulatory Guide 1.137.
11. ANSI C84.1, 1982.
12. ASME, Boiler and Pressure Vessel Code, Section XI.
13. IEEE Standard 308.
14. NUMARC 87-00, Revision 1, August 1991.

B 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 OPERABILITY of the minimum AC sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

In general, when the unit is shut down the Technical Specifications (TS) requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, and 3 have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence significantly reduced or eliminated, and minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCOs for required systems.

(continued)

BASES

LCO (continued)

powered from offsite power. An OPERABLE DG, associated with a Division 1 or Division 2 Distribution System Engineered Safety Feature (ESF) bus required OPERABLE by LCO 3.8.8, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Similarly, when the high pressure core spray (HPCS) system is required to be OPERABLE, a separate offsite circuit to the Division 3 Class 1E onsite electrical power distribution subsystem, or an OPERABLE Division 3 DG, ensure an additional source of power for the HPCS. This additional source for Division 3 is not necessarily required to be connected to be OPERABLE. Either the circuit required by LCO Item a, or a circuit required to meet LCO Item c may be connected, with the second source available for connection. Together, OPERABILITY of the required offsite circuit(s) and DG(s) ensure 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 involving handling of recently irradiated fuel, reactor vessel draindown).

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective ESF bus(es), and accepting required loads during an accident. Qualified offsite circuits are those that are described in the USAR and are part of the licensing basis for the plant. One offsite circuit consists of the Unit 1 startup transformer through the Unit 1 interbus transformer, to the Class 1E 4.16 kV ESF buses through source feeder breakers for each required division. A second acceptable offsite circuit consists of the Unit 2 startup transformer through the Unit 2 interbus transformer, to the Class 1E 4.16 kV ESF buses through source feeder breakers for each required division.

The required DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 10 seconds for Division 1 and 2 and 13 seconds for Division 3. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot and DG in standby

(continued)

BASES

LCO (continued)

with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. In addition, proper load sequence operation is an integral part of offsite circuit and DG OPERABILITY since its inoperability impacts the ability to start and maintain energized loads required OPERABLE by LCO 3.8.8. It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required AC electrical power distribution subsystems.

As described in Applicable Safety Analyses, in the event of an accident during shutdown, the TS are designed to maintain the plant in a condition such that, even with a single failure, the plant will not be in immediate difficulty.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the AC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

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BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1

A required offsite circuit is considered inoperable if no qualified circuit is supplying power to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.8, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, movement of recently irradiated fuel, and operations with a potential for draining the reactor vessel.

By allowing the option to declare required features inoperable which are not powered from offsite power, appropriate restrictions can be implemented in accordance with the required feature(s) LCOs' ACTIONS. Required features remaining powered from offsite power (even though that circuit may be inoperable due to failing to power other features) are not declared inoperable by this Required Action.

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

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building, and operations with a potential for draining the reactor vessel. Additionally, crane operations over the spent fuel storage pool shall be suspended when fuel assemblies are stored there.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to initiate

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the DC electrical power subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 4 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

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 charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7 day Frequency is consistent with manufacturer's recommendations and IEEE-450 (Ref. 8).

SR 3.8.4.2

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

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.

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B 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

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

One DC electrical power subsystem (consisting of either the Unit 1 or 2 battery, either the normal or reserve battery charger, and all the associated control equipment and interconnecting cabling supplying power to the associated

(continued)

BASES

LCO (continued)

bus), associated with the Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem(s) required OPERABLE by LCO 3.8.8, "Distribution Systems-Shutdown," is required to be OPERABLE. Similarly, when the High Pressure Core Spray (HPCS) System is required to be OPERABLE, the Division 3 DC electrical power subsystem associated with the Division 3 onsite Class 1E DC electrical power distribution subsystem required OPERABLE by LCO 3.8.8 is required to be OPERABLE. In addition to the preceding subsystems required to be OPERABLE, a Class 1E battery or battery charger and the associated control equipment and interconnecting cabling capable of supplying power to the remaining Division 1 or Division 2 onsite Class 1E DC electrical power distribution subsystem, when portions of both Division 1 and Division 2 DC electrical power distribution subsystems are required to be OPERABLE by LCO 3.8.8. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel and inadvertent reactor vessel draindown).

Division 1 consists of :
1. 125 volt battery 1R42-S002 or 2R42-S002.
2. 125 volt full capacity charger 1R42-S006 or OR42-S007.

Division 2 consists of:
1. 125 volt battery 1R42-S003 or 2R42-S003.
2. 125 volt full capacity charger 1R42-S008 or OR42-S009.

Division 3 consists of:
1. 125 volt battery 1E22-S005 or 2E22-S005.
2. 125 volt full capacity charger 1E22-S006 or OR42-S011.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment and fuel handling building provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

(continued)

BASES

APPLICABILITY (continued)	b.	Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the DC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);

(continued)

BASES

APPLICABILITY (continued)

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC subsystems remaining OPERABLE with one or more DC power sources inoperable may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, movement of recently irradiated fuel, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable DC power source(s) inoperable, appropriate restrictions are implemented in accordance with the Required Actions of the LCOs for these associated required features. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative alternate actions (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building, and operations with a potential for draining of the reactor vessel) is made.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems – Shutdown

BASES

BACKGROUND	A description of the AC and DC electrical power distribution systems is provided in the Bases for LCO 3.8.7, "Distribution Systems – Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. The OPERABILITY of the AC and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY. The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building ensures that: a. The facility can be maintained in the shutdown or refueling condition for extended periods; b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous seven days. The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the AC and DC electrical power distribution systems necessary to support OPERABILITY of Technical Specifications' required systems, equipment, and components – both specifically addressed by their own LCOs, and implicitly required by the definition of OPERABILITY.

Maintaining these portions of the AC and DC electrical power distribution systems 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 involving handling of recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 and during movement of recently irradiated fuel assemblies in the primary containment or fuel handling building provide assurance that:

a. Required features needed to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

b. Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the Distribution Systems when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous seven days);

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

(continued)

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

(continued)

BASES (continued)

ACTIONS

The ACTIONS are modified by a Note indicating that LCO 3.0.3 does not apply. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of recently irradiated fuel assemblies is not sufficient reason to require reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, movement of recently irradiated fuel, and operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the Required Actions of the LCOs for these associated required features. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of recently irradiated fuel assemblies in the primary containment and fuel handling building and operations with a potential for draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal – shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS

(continued)